December 12, 2001

Mr. L. W. Myers
Senior Vice President
Post Office Box 4
FirstEnergy Nuclear Operating Company
Shippingport, Pennsylvania 15077

SUBJECT: BEAVER VALLEY POWER STATION - NRC INSPECTION REPORT 50-334/01-09, 50-412/01-09

Dear Mr. Myers:

On November 10, 2001, the NRC completed an inspection at your Beaver Valley Units 1 & 2. The enclosed report documents the inspection findings which were discussed on November 20, 2001 with you and members of your staff.

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

Based on the results of this inspection, the inspectors identified two issues of very low safety significance (Green). Both of these issues were determined to involve violations of NRC requirements. However, because of their low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as Non-Cited violations, in accordance with Section VI.A of the NRC's Enforcement Policy. If you deny these Non-Cited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, 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 the Beaver Valley facility.

Since September 11, 2001, Beaver Valley Power Station has assumed a heightened level of security based on a series of threat advisories issued by the NRC. Although the NRC is not aware of any specific threat against nuclear facilities, the heightened level of security was recommended for all nuclear power plants and is being maintained due to the uncertainty about the possibility of additional terrorist attacks. The steps recommended by the NRC include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with local law enforcement and military authorities, and limited access of personnel and vehicles to the site.

The NRC continues to interact with the Intelligence Community and to communicate information to FirstEnergy Nuclear Operating Company. In addition, the NRC has monitored maintenance and other activities which could relate to the site's security posture.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures 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 Web site at http://www.nrc.gov/reading-rm.html (the Public Electronic Reading Room).

Sincerely,

/RA/

John F. Rogge, Chief
Projects Branch No. 7
Division of Reactor Projects

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

Enclosure: Inspection Report 50-334/01-09; 50-412/01-09
Attachment 1: 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.	50-334/01-09, 50-412/01-09
Licensee:	FirstEnergy Nuclear Operating Company
Facility:	Beaver Valley Power Station, Units 1 and 2
Location:	Post Office Box 4 Shippingport, PA 15077
Dates:	September 30 - November 10, 2001
Inspectors:	D. Kern, Senior Resident Inspector G. Wertz, Resident Inspector N. Perry, Senior Project Engineer S. Pindale, Reactor Systems Specialist D. Silk, Senior Emergency Preparedness Specialist
Approved by:	J. Rogge, Chief, Projects Branch 7 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000334-01-09, IR 05000412-01-09, on 09/30 -11/10/2001; FirstEnergy Nuclear Operating Company; Beaver Valley Power Station; Units 1 & 2. Operability Evaluations and Refueling and Outage Activities.

The inspection was conducted by resident inspectors, a regional systems engineering inspector, a regional emergency preparedness inspector, and a regional projects inspector. The inspection identified two Green findings, both of which were Non-Cited violations (NCV). The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 "Significance Determination Process" (SDP). Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violation.

A. Inspector Identified Findings

Cornerstone: Initiating Events

- **Green.** The inspectors identified a Non-Cited Violation of Technical Specification (TS) 6.8.1 for failure to follow a procedure to verify plant configuration before manipulating a valve which resulted in an unexpected discharge of borated water into the reactor. Control room operators did not adequately verify all potential flow paths prior to opening a valve (MOV-SI-863A), which resulted in approximately 600 gallons of borated water being discharged into the reactor vessel. As a result, the reactor plant remained in a higher risk configuration (reduced reactor coolant inventory and time to boil) for an additional 24 hours. Further, system restoration following this human performance error resulted in additional personnel radiation exposure (approximately 1.2 man-rem).

  This finding was of very low safety significance because all systems providing core cooling remained operable and reactor criticality was not challenged. (Section 1R20)

Cornerstone: Mitigating Systems

- **Green** The inspectors identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for failure to adequately prescribe vendor specified Auxiliary Feedwater Pump turbine bearing oil level requirements in plant procedures. This condition could result in inadequate oil lubrication to the turbine bearing and an increase in plant risk due to eventual unavailability of the Auxiliary Feedwater Pump.

  This finding was of very low significance because the Auxiliary Feedwater Pump oil level was found to be at the appropriate level and the pump was not inoperable. (Section 1R15)

B. Licensee Identified Violations

A violation of very low significance which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee were reasonable. This violation is listed in Section 4OA7 of this report.

Report Details

SUMMARY OF PLANT STATUS: Unit 1 began this inspection period in Mode 5 (cold shutdown) with a refueling outage in progress. Operators restarted the reactor on October 8, and on October 9, synchronized the unit to the electrical distribution grid following completion of a 38 day refueling outage. The NRC approved a 1.4 percent power uprate during the refueling outage. The unit achieved full rated power (2689 megawatts thermal (MWT)) on October 20. On November 6, an automatic reactor trip occurred due to a failed feedwater control valve controller (Section 4OA3). On November 8, prior to achieving criticality, a failed nuclear instrument power supply caused another automatic reactor trip during reactor restart. Following repair of the power supply, the unit restarted and synchronized to the grid on November 9. The unit achieved full power on November 10.

Unit 2 began this inspection period at 100 percent power (2652 MWT). On October 30, operators implemented a 1.4 percent power uprate and raised power to the new licensed limit of 2689 MWT. The unit remained at full power through the end of the inspection period.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignments

.1 Unit 1 Reactor Plant Component Cooling Water System

a. Inspection Scope

The inspectors performed a complete system walkdown of the Unit 1 reactor plant component cooling water (CCW) system. The inspectors reviewed Operating Manual (OM) 15, OM Figures 15-1 through 15-5, and normal system alignment checklists 1OM-15.3.B.1 and 1OM-15.3.B.2 to determine proper equipment alignments. In addition, the inspectors reviewed and evaluated the potential impact on reactor plant CCW system operation for open work orders, design change packages, engineering evaluations, and corrective action program condition reports (CRs), and reviewed the associated system health report.

b. Findings

No findings of significance were identified.

.2 Unit 1 Turbine Driven Auxiliary Feedwater System

a. Inspection Scope

The inspectors performed a partial system walkdown of the Unit 1 Turbine Driven Auxiliary Feedwater (AFW) system. The AFW system is a risk important mitigating system for emergency decay heat removal of the reactor coolant system (RCS). The inspectors reviewed OM Figure 21-1, “Main Steam,” Rev. 12, and OM Figure 24-2, “Feedwater System,” Rev. 8, prior to performing a field verification for proper equipment alignment. The inspectors observed various AFW control room indicators and reviewed the system alignment with the control room operators in order to verify as-found field conditions. Minor housekeeping items were discussed with the Nuclear Shift Supervisor (NSS).

b. Findings

No findings of significance were identified.

.3 Unit 2 Quench Spray System

a. Inspection Scope

The inspectors performed a partial system walkdown of the Unit 2 Quench Spray (QS) System. The QS system is a risk important mitigating system for containment pressure control and emergency decay heat removal of the RCS. The inspectors reviewed OM Figure 13-2, “Quench Spray,” Rev. 9, prior to performing a field verification for proper equipment alignment. The inspectors observed various QS system control room indicators and reviewed the system alignment with the control room operators in order to verify as-found field conditions. Minor housekeeping items were discussed with the NSS.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

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 risk significant areas:

- Unit 1 Intake Structure Including Diesel and Electric Fire Pumps (Fire Areas IS-1 to IS-4)
- Unit 2 Primary Auxiliary Building 735' Elevation (Fire Area PA-3)
- Unit 2 Cable Spreading Room (Fire Area CB-2)

The inspectors reviewed the fire protection conditions of the above listed areas in accordance with the criteria delineated in Nuclear Power Division Administrative Procedure (NPDAP) 3.5, “Fire Protection,” Rev. 15. Control of transient combustibles, material condition of fire protection equipment, and the adequacy of any fire protection impairments and compensatory measures were included in these plant specific reviews.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR), the Individual Plant Examination, and Individual Plant Examination of External Events to evaluate the design basis and risk significance for internal and external floods. During this inspection, the inspectors focused on the material condition of underground electrical cables. The inspectors reviewed the station's procedure for cable inspection, 1/2 MI-75-MANHOLE-1E, "Inspection of Manholes for Water Induced Damage," Rev. 3, and inspected manholes 1EMH-8A and 1EMH-8B which contained safety-related 4160 volt (V) and 480V electrical cables to the river water and service water components located in the river water intake structure. The inspectors determined that cables located in these two manholes had been essentially submerged for the last year due to groundwater intrusion from the Ohio River. Although no visible degradation was noticed following a detailed inspection of the two manholes (cabling, conduits, and supports), the inspectors discussed the submerged cable issue with the station's cognizant electrical design engineer.

Since the 4160V cables represented the highest voltage stress (volts per insulation thickness) and would therefore deteriorate faster than lower voltage cables, the inspectors focused their review on the results of insulation resistance testing done in accordance with 1/2-PMP-E-75-001, "4160 VAC Motor Inspection and Lubrication," Rev. 2. These tests, performed on a 3-year frequency, demonstrated adequate insulation resistance values (greater than 5.2 megohms) for the safety-related 4160V cables.

The inspectors also reviewed the ground fault instrumentation which would alert the control room operators to a 480V cable fault. The inspectors reviewed the Unit 1 and 2 480V emergency bus ground fault annunciation procedures: 1OM-37.4.N, "480V System Ground Isolation," Rev. 0; and, 2OM-37.4E, "Isolating a Ground on 480 VAC Busses," Rev. 3.

The inspectors reviewed the following additional documents in order to determine the acceptability of the submerged electrical cables:

- Safety Evaluation Report No. 980012-3-1, "Environmental Qualification for Submergence of Cables and Splices Installed at Beaver Valley Power Station Using the IR Trend Method."
- 1/2-ADM-2014, "Environmental Qualification of EEQML (Electrical Equipment Qualification Master List) Equipment," Rev. 0.
- 1/2-PMP-E-75-202, "480 VAC Motor Inspection, Lubrication, and Linestarter Inspection," Rev. 5.
- Engineering Memorandum 62861, Manhole Drainage.
- Drawing 8700-RE-32A, "Ductline Plan and Details, Sheet 1," Rev. 16.
- Drawing 8700-RE-100A-8, "4KV Station Service System," Rev. 8.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed Unit 1 and Unit 2 licensed operator requalification training at the control room simulators. The inspectors reviewed the operators' ability to correctly evaluate the simulator training scenario and implement the emergency plan. The inspectors observed the operators' simulator drill performance and compared it to the criteria listed in simulator scenarios listed below.

- "Unit 1 Annual Exam Drill #18," Rev. 0
- "Unit 1 Annual Exam Drill #31," Rev. 0
- "Unit 2 Annual Exam Drill #29," Rev. 0
- "Unit 2 Annual Exam Drill #33," Rev. 0

The inspectors observed supervisory oversight, command and control, communication practices, and crew assignments to ensure they were consistent with normal control room activities. The inspectors observed the response of the operators during the simulator drill transient and verified the fidelity of the simulator to the actual plant. The inspectors observed the effect training evaluators had in recognizing and correcting individual and operating crew mistakes including post-training remediation actions. The inspectors attended the post-drill critique in order to evaluate the effectiveness of problem identification. The inspectors discussed observations regarding procedure quality and crew performance with the training evaluators.

b. Findings

No findings of significance were identified

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors evaluated Maintenance Rule (MR) implementation for the issues listed below. Specific attributes reviewed included 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 Code of Federal Regulations (CFR) 50.65, "Requirements for Monitoring the Effectiveness of Maintenance of Nuclear Power Plants," and System and Performance Engineering Administrative Manual 3.2, "Maintenance Rule Program Administration," Rev. 3. The following were reviewed:

- On July 30, 2001, maintenance was completed on the Unit 1 'C' high head safety injection pump (HHSI). The pump was unavailable for about 178 hours for the work, which included replacing the speed changer and the pump motor. The 'C' pump was the installed spare, and the remaining two HHSI pumps were available during the work activities. The inspectors reviewed the MR unavailability goals as described in the licensee's MR program, and determined that no performance goals were exceeded.

- In June 2001, maintenance was completed on the Unit 2 service water pump 2SWS-P21A after its motor failed during testing on May 17, 2001. The pump was considered a spare pump for the duration of the maintenance, and accumulated about 569 hours of unavailability to complete the motor replacement activities. The inspectors reviewed the MR unavailability goals as described in the licensee's MR program, and determined that no performance goals were exceeded.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the scheduling and control of maintenance activities in order to evaluate the effect on plant risk. This review was against criteria contained in NPDAP 7.12, "Non-outage Planning, Scheduling, and Risk Assessment," Rev. 11. The inspectors reviewed the routine planned maintenance and emergent work for the following equipment removed from service:

- Emergent Repair of Unit 2 Power Range Nuclear Instrument N44 Power Supply

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed an operability evaluation in order to determine whether a proper operability justification was performed for the following item. In addition to personnel interviews and record reviews, the inspectors evaluated critical equipment parameters as specified in the associated vendor technical manual:

- On October 11, a Unit 1 plant operator questioned whether the turbine driven AFW pump outboard bearing oil level and the governor speed setting were adequate to support operability. The turbine outboard bearing oil level reading had been difficult to obtain for several shifts because the level was at the very bottom of the sightglass. Then, on October 18, the level returned to a normal indication of approximately 1/4 of an inch above the bottom of the sightglass. Operators questioned the change in level since they were unaware of any scheduled work on the AFW turbine. Prior to the refueling outage, the governor speed setting was 181 units, and no adjustment was authorized during the outage. The operators' daily log indicated that the expected setting was 181. The plant operator noticed that the actual governor setting was 179, although other plant tour operators had recorded the setting to be 181 since the plant restarted from the recent refueling outage. The operators questioned which setting was correct. On October 23, the plant operator raised both issues again during the control room shift briefing, since he had not been briefed on their resolution. The inspectors discussed these issues with the equipment operators, system engineers, and operations management in order to assess the operability of the pump and evaluate timeliness of problem resolution.

b. Findings

The inspectors identified an NCV for inadequate procedures in that a critical operational parameter for the Unit 1 turbine driven AFW pump was not properly controlled or specified in plant procedures. The inspectors determined that neither issue resulted in AFW pump inoperability since the outboard bearing oil level was measured and found to be in the correct range and the governor setting (179) was verified to have been the as-found setting prior to satisfactory performance of 1OST-24.9, "Turbine Driven AFW Pump [1FW-P-2] Operability Test," Rev. 24, performed on October 6.

The inspectors identified that the turbine outboard bearing oil sightglass had no graduations (markings) for high or low oil level as specified in vendor technical manual 8700-02.018-0001, "Terry Steam Turbine Manual," Rev. P. Additionally, 1OM-54.3.PAB1, "PAB Log Readings," Rev. 16, and 1OST-24.9, "Turbine Driven AFW Pump [1FW-P-2] Operability Test," Rev. 24, only specified that equipment operators verify the presence of oil in the sightglass (without any specification for proper level). The inspectors also noted that the oil sightglass installation configuration had been modified from the original design. The current configuration added a piping elbow which could allow the sightglass to move in the vertical direction, and thereby provide a false level indication.

The inspectors discussed these issues with system engineers who reviewed the vendor technical manual and indicated that the bearing oil level should be between 2.625 and 3 inches below the centerline of the turbine shaft. Too much or too little oil would result in insufficient operation of the oil slinger ring and damage to the bearing. Verification of proper turbine bearing operation is performed locally by the plant operators through use of the sightglass and oil pump discharge pressure gauge since neither the turbine bearing temperature nor the oil pressure readings are instrumented to alert control room operators to abnormal operation. The inspectors verified that the oil level was within the correct range by observing the system engineer measure both bearings oil level. The inboard bearing was 2.75 inches and the outboard was 2.625 inches. The inspectors determined that plant staff response to the AFW bearing oil level issues from October 11 to October 23 was untimely. Additionally, the plant staff did not recognize related operator awareness, design control, and log keeping deficiencies until identified by the inspectors. Condition Reports 01-6834, 01-7081, 01-7119, 01-7171, and 01-7279 were written to address these concerns.

The issue had a credible impact on safety, in that failure to verify and maintain correct AFW turbine bearing oil levels could result in bearing damage during operation and subsequent failure of the AFW pump. This would increase the plant risk and result in an actual impact to plant safety. The inspectors reviewed this issue in accordance with Phase 1 of the SDP and determined that the safety significance of this finding was very low (Green). The oil level was found to be at the appropriate level and therefore, the AFW pump remained operable and did not represent a loss of safety function.

10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures and Drawings,” requires that activities affecting quality shall be prescribed by documented procedures and accomplished in accordance with those procedures. Additionally, procedures shall include appropriate quantitative criteria for determining that important activities have been satisfactorily accomplished. Contrary to this requirement, vendor specified oil levels were not adequately prescribed nor verified by plant procedures. Procedure 1OM-54.3.PAB1, “PAB Log Readings,” Rev. 16, used to verify the adequacy of the AFW pump during standby condition, and 1OST-24.9, “Turbine Driven AFW Pump [1FW-P-2] Operability Test,” Rev. 24, used for operability testing did not contain specific design information to verify proper AFW turbine bearing oil level. This violation of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures and Drawings,” is being treated as an NCV consistent with Section VI.A for the NRC Enforcement Policy. This issue was entered into the corrective action program as CR 01-6834, CR 01-7081, CR 01-7119, CR 01-7171, CR 01-7242, and CR 01-7279. **(NCV 50-334/01-09-01)**

1R19 <u>Post-Maintenance Testing</u>

a. <u>Inspection Scope</u>

The inspectors reviewed and/or observed several 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 reviewed and/or observed:

- 2OST-30.6B, “Service Water Pump [2SWS*P21C] Test on Train B Header,” Rev. 3, following repair of the motor due to a failure in the winding. The inspectors reviewed the performance data obtained to ensure adequacy of the PMT.

- 1OST-24.13, “Overspeed Trip Test of Turbine Driven AFW Pump [1FW-P-2],” Rev. 6, 1OST-24.4, “Steam Turbine Driven Auxiliary Feed Pump Test [1FW-P-2],” Rev. 19, and 1OST-24.9, “Turbine Driven AFW Pump [1FW-P-2] Operability Test,” Rev. 24, following pump overhaul, seal modification, and steam supply trip valve corrective maintenance. An oil leak and thrust bearing damage occurred during the PMT due to pump assembly errors at the vendor facility. Mechanics replaced the thrust bearing shoes and reset the balance drum clearance. Following these repairs, 1OST-24.4 was successfully performed at the PMT and the pump was declared operable.

b. <u>Findings</u>

No findings of significance were identified.

1R20 <u>Refueling and Outage Activities</u>

a. Inspection Scope

The inspectors observed selected Unit 1 outage maintenance and reactor startup activities to determine whether shutdown safety functions (e.g., reactor decay heat removal, reactivity control, electrical power availability, reactor coolant inventory, spent fuel cooling, and containment integrity) were properly maintained as required by technical specifications (TSs) and license conditions. Specific performance attributes evaluated, included configuration management, communications, instrumentation accuracy, and identification and resolution of problems. The inspectors closely evaluated configuration and inventory control during periods of reduced RCS inventory due to the associated increase in shutdown risk. Specific activities evaluated included:

- 1OM-50.4.L "Reactor Startup from Mode 5 to Mode 3," Rev. 3
- 1OM-50.4.D "Reactor Startup from Mode 3 to Mode 2," Rev. 38
- 1OM-52.4.A "Raising Power from 5% to Full Load Operation," Rev. 39
- 1R14 Work Scope Add/Drop Log and work scope adjustments made in accordance with FENOC Refueling Outage Management Guideline dated April 24, 2001, and Outage Scheduling Desktop Guide (OMS-001), Rev. 5
- 1OM-6.4.N, "Draining the RCS for Refueling," Rev. 14 (operators lowered the water level in the reactor vessel head in preparation for repair activities for a leaking reactor vessel head penetration seal)
- On September 24, control room operators opened MOV-SI-863A, the Low Head Safety Injection (LHSI) to Charging Pump Suction Valve, for post-maintenance testing. Prior to this activity, reactor coolant level had been lowered to below the reactor vessel flange in preparation for reassembling the reactor head. When MOV-SI-863A was opened, the reactor vessel level immediately increased, overflowed the reactor vessel flange and spilled onto the reactor cavity floor. The inspectors reviewed the event focusing on the root cause and configuration control management.

b. Findings

Unexpected Reactor Level Excursion During Period of Reduced Reactor Coolant Inventory

] The inspectors identified an NCV of TS 6.8.1 for failure to follow a procedure, adequately verify plant configuration, and evaluate the affect of intended operations on the reactor before manipulating a valve which resulted in an unexpected discharge of borated water into the reactor. This finding is of very low safety significance because the systems providing core cooling remained operable and the excess reactor coolant did not challenge reactor criticality margins nor result in any RCS dilution.

The inspectors determined that control room operators failed to comply with 1/2OM-48.1.A, “Duties and Responsibilities of the Operations Group,” Rev. 1 when deciding to manipulate MOV-SI-863A for post-maintenance testing. The procedure specifies that the NSS shall be continuously aware of equipment condition and intended operations which could affect the reactor. The inspectors determined that control room operators had not effectively questioned the impact that opening MOV-SI-863A would have on the reactor, as they had not adequately verified all potential flow paths. As a result, a flow path from the volume control tank through the common suction of the HHSI pumps to the discharge of the low head safety injection (LHSI) pumps was created. Approximately 600 gallons of borated water discharged into the reactor vessel, some overflowing the vessel flange into the reactor cavity. Operators delayed reactor vessel head installation and RCS fill and vent to permit personnel to clean up the boric acid residue and thereby reduce the likelihood of accelerated corrosion. As a result, the plant remained in a higher risk configuration, with reactor vessel level below the reactor flange and a reduced time to boil (approximately 75 minutes), for an additional 24 hours. Additionally, workers cleaning the spilled coolant from the reactor flange and cavity received an additional 1.2 man-rem of radiation exposure.

The finding is more than minor because the plant was in a higher shut-down risk configuration (reduced time to boil) for an additional 24 hours and the additional personnel dose received resulted in an actual impact on plant safety. Phase 1 of the SDP directed the inspectors to Appendix G, “Shutdown Operations SDP.” The inspectors referred to Appendix G, Table 1, “Pressurized Water Reactor Cold Shutdown and Refueling operation and RCS open and Refueling Cavity level < 23 feet with time to boiling less than 2 hours.” The inspectors determined that controls to preclude perturbations in RCS level as addressed in Section II.B(3) of Table 1 were inadequate. The finding did not warrant a Phase 2 analysis because the event did not cause a loss of inventory, and alternate core cooling paths remained available. The safety significance of this finding was very low (Green), because all safety systems remained operable, the excess coolant added did not result in a RCS dilution, and no individual worker dose limits were exceeded.

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 written procedures are followed for operation of safety-related systems. Contrary to these requirements, operators manipulated MOV-SI-863A without adequately verifying the intended effect on the reactor as described in 1/2OM-48.1.A, "Duties and Responsibilities of the Operations Group," Rev. 1. 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 50-334/01-09-02)**. This violation was entered into the licensee's corrective action program as CR 01-3151.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed and reviewed the following operational surveillance test (OST), concentrating on verification of the adequacy of the test to demonstrate the operability of the required system or component safety function.

- 2OST-13.1, "Quench Spray Pump [2QSS*P21A] Test," Rev. 16.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

The inspectors conducted an in-office review of licensee submitted changes for the emergency plan-related documents listed below to determine if the changes decreased the effectiveness of the plan. A thorough review was conducted of documents related to the risk significant planning standards (RSPS), such as classifications, notifications, and protective action recommendations. A cursory review was conducted for non-RSPS documents. The submitted and reviewed documents (Plan and Implementing Procedures) follow:

Plan, Section 1, Definitions, Rev. 13
Plan, Section 5, Emergency Organization, Rev. 16
Plan, Section 6, Emergency Measures, Rev. 15
Plan, Section 7, Emergency Facilities and Equipment, Rev. 15
Plan, Section 8, Maintaining Emergency Preparedness, Rev. 15
Plan, Appendix G, References, Rev. 4
Emergency Preparedness Plan/Implementing Procedure (EPP/IP) I-2, Unusual Event, Rev. 16
EPP/IP I-3, Alert, Rev. 16
EPP/IP I-4, Site Area Emergency, Rev. 16
EPP/IP I-5, General Emergency, Rev. 17
EPP/IP 1.1, Notifications, Rev. 27
EPP/IP 1.2, Communication and Dissemination of Information, Rev. 17
EPP/IP 1.3, Turnover Status Checklist ED/ERM, Rev. 9
EPP/IP 1.4, Technical Support Center Activation, Operation and Deactivation, Rev. 15
EPP/IP1.5, Emergency Support Center Activation, Operation and Deactivation, Rev. 13

EPP/IP 1.6, Emergency Operations Facility Activation, Operation & Deactivation, Rev. 13
EPP/IP1.7, Emergency Response Organization (ERO) Teams, Rev. 7
EPP/IP 2.1, Emergency Radiological Monitoring, Rev. 10
EPP/IP 2.2, Onsite Monitoring for Airborne Release, Rev. 10
EPP/IP 2.3, Offsite Monitoring for Airborne Release, Rev. 10
EPP/IP 2.4, Offsite Monitoring for Liquid Release, Rev. 8
EPP/IP 2.5, Emergency Environmental Monitoring, Rev. 9
EPP/IP 2.6, Environmental Assessment & Dose Projection Controlling Procedure, Rev. 14
EPP/IP2.6.1, Dose Projection - General Methods, Rev. 10
EPP/IP2.6.2, Dose Projection - ARERAS/MIDAS With UFSAR Defaults, Rev. 12
EPP/IP2.6.3, Dose Projection - ARERAS/MIDAS With Real-Time Inputs, Rev. 12
EPP/IP 2.6.4, Dose Projection - ARERAS/MIDAS With Manual Inputs, Rev. 13
EPP/IP 2.6.12,Dose Projection-ARERAS/MIDAS With Severe Accident Assessment, Rev. 9
EPP/IP 2.7.1, Liquid Release Estimate - Computer Method, Rev. 9
EPP/IP 7.1, Emergency Equipment Inventory and Maintenance Procedure, Rev. 13

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

.1 Safety System Functional Failures

a. Inspection Scope

The inspectors reviewed the Unit 1 and Unit 2 performance indicators for safety system functional failures to determine whether the NRC approved guidance, provided in Nuclear Energy Institute (NEI) 99-02, “Regulatory Assessment Performance Indicator Guideline,” Rev. 1, was properly implemented. Verification included review of the data collected, performance indicator definitions, data reporting elements, calculational methods, definition of terms, and use of clarifying notes. The inspectors verified accuracy of the reported data, through reviews of Licensee Event Reports submitted during the period November 2000 through October 2001.

b. Findings

No findings of significance were identified.

.2 Residual Heat Removal System Unavailability

a. Inspection Scope

The inspectors reviewed the Unit 1 and 2 performance indicators for the systems that provide post-accident recirculation and shutdown cooling. The specific systems reviewed included the Unit 1 LHSI, recirculation spray, residual heat removal systems and the Unit 2 recirculation spray and residual heat removal systems. Due to the plant specific design, NEI 99-02, Appendix D, "Plant Specific Design Issues," Rev. 1, was used to determine the scope of the data collected. The inspectors verified accuracy of the reported data through reviews of the last 8 months of reported data, shift technical advisors' logs, and the December 2000 shift operator logs. In addition, the following procedures were reviewed to evaluate determination of availability.

- 2OST-11.1 "Low Head Safety Injection Pump [2SIS-P21A] Test," Rev. 16
- 2OST-11.7 "ECCS Flow Path and Valve Position Check - Train B," Rev. 11
- 2OST-11.10 "Boron Injection Flow Path Power-Operated Valve Exercise Mode 1-3," Rev. 9
- 2OST-13.5 "Recirculation Spray Pump [2RSS-P21C] Dry Test," Rev. 8
- 2OST-13.8 "Containment Depressurization System Position Verification Test-Train A," Rev. 4

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

The inspectors identified that the licensee was slow to resolve anomalous Unit 1 AFW pump speed control governor and outboard bearing oil level indications. Additionally, the licensee did not address related operator awareness, design control, and log keeping deficiencies until identified by the inspectors (Section 1R15).

4OA3 Event Follow-up

a. Inspection Scope

The inspectors evaluated two Unit 1 automatic reactor trips. On November 6, 2001, at 2:16 p.m., Unit 1 automatically tripped from 100 percent power. A failed controller caused feedwater control valve FCV-1FW-498 to close, resulting in rapidly lowering 'C' steam generator (SG) level and a reactor trip (CR 01-7371) due to low-low SG level. The inspectors responded to the control room to evaluate plant equipment and mitigating system response to the trip, operator actions including communications and use of correct emergency operating procedures, and plant stabilization to a safe shutdown condition. The inspectors observed operator actions, reviewed various instruments and sequence of events recorders, and conducted interviews to verify safe plant conditions. The inspectors also verified the reactor trip was properly reported in accordance with 10 CFR 50.72. Immediately following plant stabilization the inspectors reviewed the event's risk significance with licensee risk analysts and the NRC regional senior risk analyst. The inspectors determined that the conditional core damage probability was very low and that no additional NRC reactive response was necessary.

The inspectors attended the Unit 1 Readiness for Restart Assessment Meeting and monitored various equipment repair activities to determine whether station personnel properly evaluated plant readiness for safe restart in accordance with NPDAP 5.11, “Post-Trip Review,” Rev. 4. The Event Review Team (ERT) concluded that the apparent cause of the reactor trip was the end of life failure of a diode within a 7100 series control module. This failure caused FCV-1FW-498 to close, ‘C’ SG level to lower, and an automatic reactor trip. The inspectors determined that adequate measures were implemented to preclude repetitive challenges to safety-related equipment upon restart, as required by NPDAP 5.11.

Operators restarted the unit on November 8. Prior to achieving criticality, a fuse in the N36 intermediate range nuclear flux instrument failed and caused a high neutron flux automatic reactor trip (CR 01-7417). All control rods fully inserted as designed and the plant remained stable in Mode 3 (Hot Standby). The inspectors determined that the risk associated with this trip was very low and that no additional NRC reactive response was necessary. The ERT determined that the apparent cause of the reactor trip was the end of life failure of a capacitor within the high voltage power supply of the N36 nuclear instrument drawer. The inspectors determined that adequate measures were implemented to preclude repetitive challenges to safety-related equipment upon restart, as required by NPDAP 5.11. Operators successfully restarted the unit and synchronized the turbine to the off-site power grid on November 9.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

The inspectors presented the inspection results to Mr. Robert Saunders, Mr. Lew Myers, and other members of licensee management following the conclusion of the inspection on November 20, 2001. The licensee acknowledged the findings presented.

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

4OA7 Licensee-Identified Violations

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

Requirement Licensee Failed to Meet

Technical Specification 3.6.3.1 requires each containment isolation valve to be operable. If a containment isolation valve is not operable, TS action statement 3.6.3.1.b. requires the affected penetration to be isolated within 4 hours. Contrary to the above, on October 22, 2001, at 1:16 p.m., pressurizer relief tank gas sample inside containment isolation valve, 2SSR-SOV130A1, closed abnormally slowly during containment isolation valve testing. Control room operators performing the surveillance test did not recognize that the valve was inoperable, and therefore failed to isolate the penetration within 4 hours. The valve was declared inoperable at 5:10 p.m. following a review of the test data by the relieving NSS. The containment penetration was successfully isolated at 7:06 p.m. Condition Report 01-7069 was initiated to enter the issue into the corrective action program. **(NCV 50-334/01-09-03)**

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

a. Key Points of Contact

T. Cosgrove	Manager, Nuclear Regulatory Affairs
R. Donnellon	Director, Projects and Scheduling
R. Fast	Director, Plant Maintenance
L. Myers	Senior Vice President, FENOC
M. Pearson	Director, Nuclear Services
F. von Ahn	Director, Plant Engineering
L. W. Pearce	Plant General Manager

b. Items Opened, Closed and Discussed

Opened and Closed

50-334/01-09-01	NCV	Failure to Prescribe and Verify Auxiliary Feedwater Pump Turbine Oil Level Requirements in Plant Procedures (Section 1R15)
50-334/01-09-02	NCV	Failure to Evaluate the Affect of Intended Operations on the Reactor Resulted in a Reactor Coolant Spill (Section 1R20)
50-334/01-09-03	NCV	Failure to Isolate Pressurizer Relief Tank Gas Sample Line Containment Penetration Within 4 Hours of Identifying Inoperable Containment Isolation Valve (2SSR-SOV130A1) per TS 3.6.3.1.b (Section 4OA7)

c. List of Acronyms Used

ADAMS	NRC's Document System
AFW	Auxiliary Feedwater
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
EEQML	Electrical Equipment Qualification Master List
EPP/IP	Emergency Preparedness Plan Implementing Procedure
ERO	Emergency Response Organization
ERT	Event Review Team
HHSI	High Head Safety Injection
LHSI	Low Head Safety Injection
MR	Maintenance Rule
MWT	Megawatts Thermal
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute

NPDAP	Nuclear Power Division Administrative Procedure
NRC	Nuclear Regulatory Committee
NUREG	NRC Technical Report Designation (<u>Nu</u>clear <u>Reg</u>ulatory Commission)
NSS	Nuclear Shift Supervisor
OM	Operating Manual
OST	Operational Surveillance Test
PARS	Publicly Available Records
PMT	Post-Maintenance Test
QS	Quench Spray
RCS	Reactor Coolant System
RSPS	Risk Significant Planning Standard
SDP	Significance Determination Process
SG	Steam Generator
SSC	Structures, Systems, and Components
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
V	Volt