May 30, 2002

Mr. L. W. Pearce
Vice President - FENOC Oversight
& Beaver Valley Plant General Manager
Post Office Box 4
FirstEnergy Nuclear Operating Company
Shippingport, Pennsylvania 15077

SUBJECT: BEAVER VALLEY POWER STATION - NRC INSPECTION REPORT 50-334/02-04, 50-412/02-04

Dear Mr. Pearce:

On May 11, 2002, the NRC completed an inspection at your Beaver Valley Units 1 & 2. The enclosed report documents the inspection findings which were discussed with you and members of your staff on May 16, 2002.

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.

No findings of significance were found.

Immediately following the terrorist attacks on the World Trade Center and the Pentagon, the NRC issued an advisory recommending that nuclear power plant licensees go to the highest level of security, and all promptly did so. With continued uncertainty about the possibility of additional terrorist activities, the Nation's nuclear power plants remain at the highest level of security and the NRC continues to monitor the situation. This advisory was followed by additional advisories, and although the specific actions are not releasable to the public, they generally include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with law enforcement and military authorities, and more limited access of personnel and vehicles to the sites. The NRC has conducted various audits of your response to these advisories and your ability to respond to terrorist attacks with the capabilities of the current design basis threat (DBT). On February 25, 2002, the NRC issued an Order to all nuclear power plant licensees, requiring them to take certain additional interim compensatory measures to address the generalized high-level threat environment. With the issuance of the Order, we will evaluate FirstEnergy Nuclear Operating Company's compliance with these interim requirements.

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,

/NA/

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

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

Enclosures: Inspection Report 50-334/02-04; 50-412/02-04
Attachments: 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.	50-334/02-04, 50-412/02-04
Licensee:	FirstEnergy Nuclear Operating Company
Facility:	Beaver Valley Power Station, Units 1 and 2
Location:	Post Office Box 4 Shippingport, PA 15077
Dates:	March 31 - May 11, 2002
Inspectors:	D. Kern, Senior Resident Inspector S. Barr, Project Engineer G. Cranston, Reactor Inspector J. Jang, Senior Health Physicist D. Orr, Senior Resident Inspector N. Perry, Senior Project Engineer R. Bhatia, Reactor Inspector
Approved by:	J. Rogge, Chief, Projects Branch 7 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000334-02-04, IR 05000412-02-04, on 3/31-5/11/2002; FirstEnergy Nuclear Operating Company; Beaver Valley Power Station; Units 1 & 2. Resident Inspector Report.

The inspection was conducted by resident inspectors, a regional senior health physicist, and regional projects inspectors. 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. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

No significant findings were identified.

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 appear reasonable. The violation is listed in Section 4OA7 of this report.

Report Details

SUMMARY OF PLANT STATUS

Unit 1 began this inspection period at 95 percent power in order to reduce the effects of elevated ‘A’ main feedwater pump outboard motor bearing temperature. On April 2, operators performed a planned power reduction to 63 percent. Following bearing replacement and ‘C’ and ‘D’ condenser waterbox cleaning, power was restored to 100 percent on April 5. On April 12, operators reduced power to 90 percent for additional planned condenser waterbox cleaning. Power was further reduced to 80 percent to maintain adequate condenser vacuum. Full power was restored on April 14.

Unit 2 began this inspection period at 100 percent power. On April 4, operators reduced power to 90 percent for condenser waterbox cleaning and reestablished 100 percent power on April 5. Shortly after achieving full power, chemists reported that all three steam generators (SGs) had high levels of sulfates and sodium, indicating a condenser tube leak. Operators promptly initiated an unplanned shutdown to 2 percent power to minimize the corrosion effects of the adverse SG chemistry conditions (Section 1R14). Leaks in the ‘C’ condenser waterbox were repaired and power was restored to 100 percent on April 6. Unit 2 performed brief planned power reductions to 75 percent power for further condenser waterbox cleaning on April 20-21 and for additional condenser waterbox tube leak repairs on May 11.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignments

Unit 1 Supplementary Leak Collection and Release System

a. Inspection Scope

The inspectors performed a partial system alignment walkdown of the Unit 1 supplementary leak collection and release system (SLCRS) while the ‘A’ SLCRS train was out of service for planned maintenance and surveillance testing. The inspectors verified the ‘B’ SLCRS train was aligned as required by Operating Manual (OM)-16.3.A, “SLCRS System and Component Arrangement,” Rev. 1; 1OM-16.3.C, “SLCRS Power Supply and Control Switch List,” Rev. 5, and OM Figure Number 16-1, “Ventilation and Air Conditioning Primary Plant,” Rev. 11. The inspectors also verified applicable technical specification (TS) limiting conditions of operation were properly implemented while the ‘A’ SLCRS inoperable.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Fire Protection Area Inspections

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 Process Instrument Room (Fire Area CR-4)
- Unit 1 Quench Spray & Auxiliary Feedwater Subarea (Fire Area PT-1 (QP-1))

The inspectors reviewed the fire protection conditions of the above listed areas in accordance with the criteria delineated in Nuclear Power Administrative Manual, 1/2-ADM-1900, “Fire Protection,” Rev. 0. 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.

.2 (Closed) Unresolved Item (URI) 50-334/01-05-01: Inadequacy of the Hemyc Cable Wrap Fire Barrier Test and Evaluation

During the June 2001 triennial fire protection inspection at Beaver Valley Unit 1, the NRC team found that the licensee had used Hemyc fire wrap to protect raceways associated with the power supply feeder cable for CH-P-1B charging pump in the auxiliary building. The Hemyc fire wrap was utilized to meet the separation requirements of 10 Code of Federal Regulations (CFR) 50, Appendix R. These raceway enclosures required a one- hour fire rating to ensure availability of reactor coolant inventory makeup for postulated fires in the auxiliary building. Because Hemyc fire barriers were installed after the effective date of Appendix R, they were required to meet the technical requirements of Appendix R or have appropriate documentation to justify a deviation.

The team also noted the NRC had previously identified issues at Shearon Harris Nuclear Power Plant (IR 50-400/99-13) regarding the acceptability of Hemyc fire wrap qualification tests. The team determined the Hemyc fire wrap installed configuration at Beaver Valley Unit 1, supported by design analysis 8700-DEC-0187 for one-hour fire resistance capability, was also based, in part, on the results of that qualification test. The issue was left as an URI pending the licensee’s actions to address this concern.

To address the Hemyc fire wrap material concern, the licensee replaced the existing Hemyc fire wrap material with Darmatt KM-1. The inspectors’ review of design analysis 8700-DEC-0234, ”Fire Wrap Analysis for CH-P-1B Power Cable,” of the installed Darmatt configuration at Beaver Valley Unit 1 was adequately supported by tests performed by the Faverdale Technology Center Ltd. test facility. The inspectors found that the licensee had adequately analyzed the installed Darmatt configuration of raceways and supports for the CH-P-1B charging pump feeder cable. The Darmatt fire wrap was installed in fire areas PA-1E and 1G of auxiliary building at elevation 735'-6" and 722'-6" respectively. The Darmatt fire barrier installation was verified to be consistent with the design analysis and was appropriately justified in this analysis by tests to ensure one-hour fire resistance capability. Review of the licensee's analysis for feeder cable ampacity indicated that they had appropriately used reasonable ampacity and temperature derating factors. The inspectors concluded that the feeder cable in the Darmatt fire barrier configuration had adequate capacity to supply the CH-P-1B charging pump safety load.

The inspectors determined that based on the small amount of Hemyc fire wrap on the CH-P-1B charging pump cable run, the good material condition of the Hemyc fire wrap that was installed, an operable detection and automatic suppression system on the 735' elevation, low combustible loading on the 722' elevation, and an effective fire brigade, a fire in fire areas PA-1E and 1G would not have had a credible impact on plant safety. A fire in these areas would have been detected early, and fire suppression/fighting techniques would have extinguish the fire before the loss of the CH-P-1B charging pump could occur. This issue was considered minor and not subject to formal enforcement. The station personnel replaced the Hemyc fire wrap with the Darmatt fire wrap to ensure one-hour protection capability of the CH-P-1B charging pump cables. This was accomplished through the licensee's corrective action (CA) process.

Based on this review and the satisfactory field verification of the Darmatt fire barrier installation and configuration, the inspectors concluded this item is closed.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed Unit 1 licensed operator training at the control room simulator, focusing on human performance of time critical tasks. The inspectors reviewed the operators' ability to correctly evaluate the simulator training scenario, identify and perform response procedures, and implement the emergency plan. The inspectors observed the operators simulator drill performance and compared it to the criteria listed in simulator scenario drill number 38. 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. Scenario response procedures included the following:

- Abnormal Operating Procedure 1.51.1, "Emergency Shutdown," Rev. 9
- Emergency Operating Procedure (EOP) E-0, "Reactor Trip or Safety Injection," Rev. 1
- EOP E-3, "Steam Generator Tube Rupture," Rev. 1

- 1OM-53A.1.FR-S.1, “Response to Nuclear Power Generation - Anticipated Transient Without Scram,“ Rev. 2

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 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.

- Two of the four Unit 2 emergency diesel generator air start tank relief valves failed to lift at set pressure when tested (Condition Report [CR] 02-01734 and CR 02-01484). The inspectors interviewed the system engineer and reviewed plans to improve system reliability. The system was designated as a MR category (a)(2) system.

- The 21B steam generator (SG) atmospheric steam dump valve operator failed due to the failure of a refurbished power amplifier module (CR 02-01738). The inspectors interviewed the system engineer and reviewed plans to improve system reliability. The system was designated as a MR category (a)(2) system.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions

.1 Forced Unit 2 Shutdown Due to Adverse Steam Generator Chemistry

a. Inspection Scope

The inspectors reviewed human performance during the following nonroutine plant evolution, to determine whether personnel performance caused unnecessary plant risk or challenges to reactor safety.

- On April 5, 2002, shortly after Unit 2 was returned to full power operation following main condenser waterbox cleaning, chemists identified high levels of contaminants (sodium, sulfates, chlorides) in all three Unit 2 SGs. Operators attempted to identify the source of the contaminants. Station personnel subsequently determined that the SG contamination was caused by a main condenser tube leak. The leak had been caused when operators isolated the main condenser waterboxes for cleaning at a high power level (90 percent power), which created a higher mechanical stress than the condenser tubes could structurally withstand.

  The concentration of contaminants increased rapidly. Within 3 hours, SG chemistry had reached Action Level 3, as defined in Electric Power Research Institute (EPRI) Secondary Water Chemistry Guidelines, Rev. 5. Action Level 3 indicates that conditions exist which will result in rapid SG tube corrosion. The EPRI guideline recommends the plant be shut down to less than 5 percent reactor power as quickly as safe plant operation permits.

  The inspectors observed operators identify and isolate the source of the leakage, reduce power to 2 percent, and perform SG chemistry clean-up activities. The inspectors verified actions were performed as specified in 1/2OM-48.1.I, "Technical Specification Compliance," Rev. 9; 2OM-52.4.B, "Load Following," Rev. 39; 2OM-51.4.A, "Plant Shutdown from 40 percent Power to Mode 3," Rev. 13; and the EPRI chemistry guideline. The turbine was taken off-line at 2:24 p.m. and the reactor entered Mode 2 (<5 percent power) at 2:43 p.m. The plant remained at low power for 18 hours while operators configured the plant to improve SG chemistry conditions and address contaminant introduction through SG hideout return. Operators returned the plant to full power late on April 6.

  The inspectors noted that operators had halted the power reduction twice (for 31 minutes at 80 percent power and 1 hour 22 minutes at 29 percent power) and questioned the effect this had on SG tube corrosion. The EPRI guidance indicated that continued operation above 5 percent power with adverse SG chemistry conditions created an undesired effect of SG tube corrosion. Station personnel determined that their knowledge of the EPRI chemistry guidelines was deficient and initiated several CRs to address lessons learned from this event (CRs 02-2701 and 02-2727).

b. Findings

No findings of significance were identified.

.2 (Closed) Licensee Event Report (LER) 412/2001-03: Condition Inadvertently Exceeds Technical Specification Allowed Outage Time

a. Inspection Scope

On November 20, 2001, a relay crew at Unit 2 performed 2MSP-36.28-E, “21C Reactor Coolant Pump (RCP) 4 kilovolt (kV) Bus Underfrequency Relay, 81-VC200 Functional Test,” Rev. 8, to test a relay used in the reactor protection system (RPS). Unexpected indications during the test caused the test to be terminated and the system was restored to normal status. The same indications were received when the test was re-performed on November 21. Technicians determined that the cause of the unexpected indications was a failed relay, which rendered the affected channel of the RPS inoperable. Upon discovery, operators entered the appropriate TS limiting condition of operation action statement (TS 3.3.1.1, Item 17, Action Statement 7). However, due to the delay in recognizing the significance of the unexpected indications, the 6-hour allowed outage time for the condition had already been exceeded. The cause of the event was human error by operators and relay crew personnel. They failed to question and investigate the unexpected indications in a timely manner. As a result they did not identify that a RPS channel was inoperable for approximately 24 hours. The inspectors reviewed the LER, the Human Performance Evaluation, and associated corrective actions in order to assess the depth and adequacy of the licensee’s response to the event.

b. <u>Findings</u>

No findings of significance were identified.

1R15 <u>Operability Evaluations</u>

a. <u>Inspection Scope</u>

The inspectors reviewed operability evaluations in order to determine that proper operability justifications were performed for the following items. In addition, where a component was determined to be inoperable, the inspectors verified the TS limiting condition for operation implications were properly addressed.

- On April 8, 2002, control room operators received several low lube oil temperature alarms on the in-service ‘A’ charging pump. Operators responded to the alarms and identified that a lube oil temperature switch was causing the ‘A’ charging pump lube oil to prematurely receive full cooling through the associated closed cooling water heat exchanger. The inspectors reviewed CR 02-02749, “Inadequate Control of CH-P-1A Lube Oil Temperature,” walked down the ‘A’ charging pump lube oil system, and reviewed FENOC’s operability evaluation for the degraded lube oil condition.

- On April 17, the Unit 2 leading edge flow meter (LEFM) feedwater flow indication input to the plant computer system degraded to a “Red/?” unreliable status. The LEFM data input is used for the daily heat balance (or power level) calculation and associated nuclear instrument high flux trip setpoint adjustment. The inspectors reviewed the impact the unreliable LEFM data quality had on operability of the LEFM and the corresponding effect on the licensed power limit according to the License Requirements Manual, Section 3.8. Operators were aware that the licensed power limit must be reduced if the LEFM was not operable when the next daily heat balance (or power level) calculation was due.

Engineers, technicians, and vendor personnel determined that the LEFM signal quality had degraded. Raised the LEFM gain setting, which restored a sufficient signal to noise ratio to ensure reliable LEFM data quality. The LEFM was operable to perform the required daily heat balance calculation on April 18. Deficiencies identified during this activity were documented in CR 02-2990, 02-3831, and 02-3830.

- The regulator setting for air inlet pressure to Unit 1 solenoid operated valve (SOV) SOV-1SS-000B was raised from 20 pounds per square inch gauge (psig) to 30 psig, which exceeded the maximum operating pressure deferential for the SOV (CR 01-08250). The regulator provides air to nine air-to-open, fail closed, primary containment isolation air operated valves (AOV) through the SOV. The inspectors observed that the licensee had identified the problem, taken appropriate action to prevent recurrence, and completed an operability determination (CR 01-8250) which determined that the SOV, though degraded, was operable for the approximate 3-month period that the regulator was set at the higher pressure. Additionally, the higher air pressure did not exceed the design limits of the AOVs, nor affect the AOVs containment isolation capabilities since the valves are air-to-open, spring-to-close, and the AOVs met their stroke time criteria with the regulator set at 20 psig and at 30 psig. The regulator setting was restored to the proper value of 20 psig.

- The Unit 1 charging pump CH-P-1B minimum recirculation flow was at the maximum limit indicating that the recirculation flow orifice could be degraded, affecting the design flow to the reactor coolant system (RCS). The inspectors observed that the licensee had identified that the problem resulted from the increased flow from a sightly stronger charging pump that had been installed since the last test and that the CA to prevent recurrence was to revise the maximum recirculation allowable flow slightly upward while still achieving required flow to the RCS. The licensee completed an operability determination (CR 01-8451) and a calculation (Addendum 3 to 8700-DMC-3072, “Minimum Operating Performance,” Rev. 3), which determined that the charging pump was operable. Additionally, based on a trend of past and current test data, engineers determined that the recirculation line flow restricting orifice had not degraded and that adequate flow was being discharged from the charging pump to the RCS.

- A cloth diaper and leak repair injection compound were found in the Unit 2 residual heat removal system (CR 02-01221) downstream of relief valve 2RHS-RV721B. The inspectors reviewed the licensee’s operability determination, which concluded that the foreign material would not have impaired the functionality of the relief valve or the discharge piping. The inspectors reviewed the licensee’s evaluation of problems with foreign material to determine whether they were of sufficient detail and scope to identify and address the extent of the condition. Additionally, the inspectors reviewed CAs, which included issuing a new procedure (NOP-WM-4001, Foreign Material Exclusion), to determine if the CAs addressed the identified causes and had been implemented in a time frame commensurate with the safety significance of the problem.

- The inspectors reviewed the licensee's evaluation of problems with the low head safety injection pump relief valve (2SIS-RV8864A) lifting slightly (simmering) for a short duration of time during plant heat ups and cool downs (CRs 02-01739, 99-01757, and 99-02876) due to a small amount of back leakage through a check valve. The leakage allows pressure to build up in the safety injection pump line causing the relief valve to open slightly (simmer) when pressure approaches its design lift pressure. The inspectors reviewed the licensee's evaluation that the simmering was expected, was acceptable, and did not present a risk or safety problem since: (1) the check valve is not designed to be leak tight at a RCS pressure below 1000psig; (2) the check valve is periodically tested per the Inservice Testing (IST) program to ensure it meets its design basis leakage limits (2 Operational Surveillance Test [OST]-11.16, Rev. 15); (3) the simmering only occurs for a short period of time during cooldown and heat up; and, (4) the relief valve is functioning properly and is performing its intended design function of protecting the safety injection system piping.

- The inspectors reviewed the licensee's evaluation of two recent industry operating experience issues which could potentially effect operability of the high head safety injection (HHSI)/charging system. The first issue addressed HHSI/charging pump motor failures (CR 02-02161). The inspectors determined that the licensee had established a long-term task to refurbish large motors, including the HHSI/charging pump motors, and that one of the HHSI/charging pumps had been refurbished. Additionally, based on the operating experience review, the licensee had reduced the allowed time in-service criteria requiring pump refurbishment. The second issue addressed recent failures of Bailey valve positioners, similar to those used on AOVs in the feedwater, reactor coolant, and charging systems at Beaver Valley Unit 1 (CR 01-07641). For the Bailey positions used on AOVs, the inspectors observed that the licensee had conducted an operability review, had determined that the problems identified at the other plant did not exist at Beaver Valley, and had determined that there were no valve operability concerns.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed and 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 observed:

- Unit 1 Beaver Valley Test 2.30.1, "River Water Pump [1WR-P-1A] Head Capacity Curve," Rev. 10, following replacement of the pump. The inspectors observed portions of the test, and compared test results against the procedure acceptance criteria. The inspectors reviewed the preliminary engineering evaluation for the pump indicating it was operating acceptably and was capable of fulfilling its safety function. Additionally, the inspectors discussed the testing and the results with operations personnel and the IST coordinator.

- 1OST -16.1 "Supplementary Leak Collection and Release Test for Exhaust Through the Main Filter Bank - Train A," Rev. 7, following charcoal filter sampling and deluge valve testing.

- 2OST-7.5 "Centrifugal Charging Pump 2CHS-P21B," Rev. 22 following rebuild of the pump.

- 1OST-30.6A "Reactor Plant River Water Pump 1C Test on Train 'A' Header," Rev. 4, to evaluate backflow through the 1A reactor plant river water (RW) pump discharge check valve, 1RW-57, following corrective maintenance. On April 3, 2002, mechanics had identified that the 1RW-57 valve rubber seating material was missing. 1RW-57 was repaired with a new rubber seat. The inspectors also interviewed system engineers and determined that 1RW-57 valve backflow had been properly tested prior to discovery of the degraded seat. Engineers determined that the metal surfaces most likely provided an adequate seating surface. Station personnel entered this equipment problem into the CA program as CR 02-02625.

- 1OST-15.1 "'A' Reactor Plant Component Cooling Water Pump," Rev. 10, following planned maintenance.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

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

- 1OST-13.1 "Quench Spray Pump [1QS-P-1A] Test," Rev. 21

- 1OST-30.2 "Reactor Plant River Water Pump 1A Test," Rev. 25

- 1OST-30.6A "Reactor Plant River Water Pump 1C Test on Train 'A' Header," Rev. 4.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed temporary modifications (TMs) and associated implementing documents to verify the plant's design basis and effected system or component operability were maintained. Beaver Valley Power Station Administrative Procedure ½-ADM-2028, "Temporary Modifications," Rev. 0, specified requirements for development and installation of TMs. The inspectors reviewed TMs associated with the following items:

- Inspectors reviewed all Unit 2 TMs for their cumulative impact on safety and operability of safety-related equipment. There were no TMs directly installed on any Unit 2 safety-related systems. The inspectors performed a detailed examination of TM 2-02-03, "Electro-hydraulic Fluid Leak Collection," for the effect on the electro-hydraulic system reliability and the effect on initiating event frequency of a turbine trip.

- The inspectors reviewed Unit 1 TM 1-02-01, "Cooling Tower Pump House Ventilation," which blocked closed the recirculation dampers for fan VS-F-54B to prevent hot air being drawn back into the system and to block open the outside dampers to allow outside air to be drawn into the building to provide ventilation for the cooling tower water pumps. Though the cooling tower pump house ventilation system is not classified risk-significant or in the MR, loss of the system could result in loss of the cooling tower water pumps with the resultant loss of condenser vacuum and a plant trip. The inspectors verified that the TM did not affect system operability/availability. Additionally, the TM was removed during the inspection period when new, replacement damper actuators were installed. The inspectors verified that normal system operation was restored.

b. Findings

No findings of significance were identified.

Emergency Preparedness (EP)

1EP6 Drill Evaluation

.1 Unit 1 Control Room Simulator Emergency Plan Training Scenario

a. Inspection Scope

The inspectors observed an operations department training evolution conducted at the Unit 1 control room simulator to evaluate emergency procedure implementation, event classification, event notification, and protective action recommendation development. The event scenario involved multiple safety-related component failures and plant conditions warranting a simulated Alert event declaration. The licensee counted this training evolution for evaluation of Emergency Preparedness Drill/Exercise Performance (DEP) Indicators. The inspectors observed the training critique to determine whether the licensee critically evaluated operator performance to identify deficiencies and weaknesses. The inspectors reviewed the event notification forms and DEP indicator results during this period to verify the DEP performance indicators were properly evaluated consistent with Nuclear Energy Institute (NEI) 99-02, “Regulatory Assessment Performance Indicator Guideline,” Rev. 2. Additional documents used for this inspection activity included:

- Abnormal Operating Procedure 1.51.1, “Emergency Shutdown,” Rev 9
- EOP E-0, “Reactor Trip or Safety Injection,” Rev. 1
- EOP E-3, “Steam Generator Tube Rupture,” Rev. 1
- Emergency Plan Implementing Procedure (EPIP) 1.1, “Notifications,” Rev. 28
- Emergency Preparedness -16, “NRC Emergency Preparedness Performance Indicator Instructions,” Rev. 4

b. Findings

No findings of significance were identified.

.2 Unit 1 Emergency Preparedness Drill Evolution

a. Inspection Scope

The inspectors observed a Unit 1 emergency event training evolution to evaluate emergency procedure implementation, event classification, event notification, and protective action recommendation development. The Operations Support Center, Radiological Operations Center, Technical Support Center, and Emergency Operations Facility were activated and participated in this drill. The event scenario involved multiple safety-related component failures and plant conditions warranting simulated alert and site area emergency event declarations. The licensee counted this training evolution for evaluation of Emergency Preparedness DEP Indicators. The inspectors reviewed the drill critique report to determine whether the licensee critically evaluated drill performance to identify deficiencies and weaknesses. Additionally, the inspectors verified the DEP indicators were properly evaluated consistent with NEI 99-02, “Regulatory Assessment Performance Indicator Guideline,” Rev. 2. Additional documents used for this inspection activity included:

- Beaver Valley Power Station 2002 Mini Drill Scenario
- EOP E-0 “Reactor Trip or Safety Injection,” Rev. 1
- EOP E-2 “Faulted Steam Generator Isolation,” Rev. 0
- EOP E-3 “Steam Generator Tube Rupture,” Rev. 1
- Emergency Action Level (EAL) 1.2, "Reactor Coolant System Barrier,” Rev. 6
- EAL 1.3 “Containment Barrier,” Rev. 6
- EAL 4.6 “Security,” Rev. 6
- EPIP 1.1 “Notifications,” Rev. 28
- EP-16 “NRC Emergency Preparedness Program Performance Indicator

Instructions," Rev. 4

The inspectors observed that initial event notifications to the state/local agencies were not completed within 15 minutes. Additionally, the notifications incorrectly stated that no release was in progress, even though the steam driven auxiliary feedwater (AFW) pump supply valve from the faulted A' SG (MS-15) was failed open. This caused a release path to the environment. The licensee drill critique addressed these items (CRs 02-3442, 3477, 3529, and 3536).

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

CORNERSTONE: Public Radiation Safety

2PS1 Radiological Environmental Monitoring Program (REMP)

a. Inspection Scope

The inspectors reviewed the following documents to ensure the licensee met the requirements specified in the Technical Specification/Offsite Dose Calculation Manual (TS/ODCM):

- the 2000 Annual REMP Reports;
- the most recent ODCM (REMP Section) and technical justifications for ODCM changes;
- the most recent calibration results of the primary and backup meteorological monitoring instruments for wind direction, wind speed, and temperature at 33-ft, 150-ft, and 250-ft levels;
- operability of the meteorological monitoring instruments;
- the most recent calibration results for air samplers;
- implementation of the environmental thermoluminescent dosimeters (TLDs) program;
- the Land Use Census procedure and the 2000/2001 results;
- the Quality Control evaluation of the interlaboratory and intralaboratory comparison programs and the CAs for any deficiencies;
- CRs 02-00428; 02-01838; 02-02443; 01-1552; 01-3132; 01-3918; 01-3446; 01-7422; and 01-7453 and the associated CAs;
- Quality Assurance Audit (Audit Report Number BV-C-01-12) for the REMP/ODCM implementations; and
- associated REMP procedures, including vendor's analytical procedures.

The inspectors also toured and observed the following activities to evaluate the effectiveness of the REMP.

- observation for the operability of meteorological monitoring instruments located at the primary and backup towers;

- surface water sampling station (automatic water sampler); and
- visual inspection for determining whether air samplers, milk farms, and 25 percent TLDs were located as described in the ODCM (including control and indicator stations) and for determining the equipment material condition.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Control Program

a. Inspection Scope

The inspectors reviewed the following documents to ensure the licensee met the requirements, concerning the unrestricted release of material from the radiologically controlled area (RCA), specified in the licensee's program:

- the most recent calibration results for the radiation monitoring instrumentation (SAM-11), including the (a) alarm setting, (b) response to the alarm, and (c) the sensitivity;
- the criteria for the survey and release of potentially contaminated material using a gamma spectroscopy (calibration efficiency for bulk sample analyses);
- the methods used for control, survey, and release from the RCA; and
- observed monitor calibration and records to verify the lower limits of detection for bulk sample analyses.

The review was against criteria contained in 10 CFR 20, NRC Circular 81-07, "Control of Radioactive Material," NRC Information Notice 85-92, "Surveys of Water before Disposal from Nuclear Facilities," NUREG/CR-5569, "Health Position Data Base (Positions 221 and 250)," dated February 1994, and the licensee's applicable procedures.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

High Head Safety Injection and Auxiliary Feedwater Safety System Unavailability

a. Inspection Scope

The inspectors reviewed the Unit 1 and 2 performance indicators (PIs) for the HHSI and AFW systems to ensure the PIs were reported in accordance with the guidance in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Rev. 2. The inspectors verified accuracy of the reported data through reviews of the last six months (July 2001 - January 2002) of reported data, plant logs, and system availability performance reports. In addition, the following procedures were reviewed to evaluate determination of availability:

- 1OST-7.4 "Centrifugal Charging Pump Test (1CH-P-1A)," Rev. 21
- 2OST-7.5 "Centrifugal Charging Pump Test (2CHS*P21B)," Rev. 22
- 1OST-24.2 "Motor Driven Auxiliary Feed Pump Test [1FW-P-3A]," Rev. 21
- 2OST-24.4 "Steam Driven Auxiliary Feed Pump Test (2FWE*P22) Quarterly Test," Rev. 41

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed the licensee's evaluation of pipe minimum wall thickness deficiencies on new pipe that was scheduled for installation in the HHSI/charging pump systems for Units 1 & 2 (CRs 02-00398 and 01-07179).

b. Findings

The inspectors determined that, although the licensee's initial receipt inspections did not detect the minimum wall thickness minor deficiencies on a portion of the pipe, the licensee's subsequent inspections did, and the discrepant pipe did not get installed in the plant. Additionally, the licensee determined that, for the sections of pipe where the wall thickness was less then the minimum allowable, the measured pipe wall thickness was still adequate to maintain piping integrity for the rated pressure of the pipe had it been installed. The inspectors found the licensee's causal analysis, extent of condition review, and actions to prevent recurrence appropriate.

No findings of significance were identified.

4OA3 Event Follow-up

(Closed) LER 50-334/01-03: Automatic Reactor Trip Due to Low Steam Generator Water Level

This event was discussed in NRC Inspection Report Nos. 50-334(412)01-09. Causal analysis and the schedule for CA implementation were appropriate. No new issues were revealed by the LER. This LER was closed during an onsite review.

4OA6 Management Meetings

.1 Exit Meeting Summary

The inspectors presented the inspection results to Mr. L. W. Pearce and other members of licensee management following the conclusion of the inspection on May 16, 2002. The licensee acknowledged the findings presented.

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

.2 Site Management Visit

On April 9, 2002, a public meeting was held at the FirstEnergy Operating Company (FENOC) Beaver Valley site Emergency Response Facility. Mr. John Rogge, Chief, Reactor Projects Branch 7, NRC Region I, and other NRC staff members discussed the results of the NRC's assessment of safety performance at Beaver Valley Power Station for the period of April 1, 2001, to December 31, 2001. Mr. Robert Saunders, President, FENOC, Mr. Lew Myers, Chief Operating Officer, FENOC, and other members of their staff attended the meeting.

Presentation slides from the Beaver Valley Annual Assessment meeting are available on the ADAMS website http:/www.nrc.gov/reading-rm/adams.html, Accession Number ML021010147.

.3 FirstEnergy Nuclear Operating Company Senior Management Changes

On April 29, 2002, FENOC announced several interim management changes associated with Davis-Besse oversight, reactor vessel head restoration, and plant restart activities. Mr. Lew Myers, currently Senior Vice President at Beaver Valley, will provide senior oversight while assisting Davis Besse's management team. Mr. Bill Pearce, Beaver Valley General Plant Manager, will assume the additional duties of Beaver Valley Site Vice President. John Wood, Vice President of FENOC Engineering Services will be dedicated to the role of technical lead for restoration of Davis-Besse's reactor vessel head. The Engineering groups at Perry and Beaver Valley will report directly to the respective Site Vice Presidents during this interim period.

On May 8, 2002, FENOC announced three newly created management positions and several permanent senior management changes as presented below:

> Gary R. Leidich, Executive Vice President of the Institute of Nuclear Power Operations has be named to the newly created position of Executive Vice President - FENOC and will report to Mr. Saunders. Reporting to Mr. Leidich will be John Wood, Vice President of FENOC Engineering Services and directors of engineering from FENOC's Davis-Besse, Perry and Beaver Valley plants. He will also have responsibility for GPU Nuclear. His appointment is effective June 10, 2002.
>
> Lew W. Meyers, FENOC Senior Vice President at the Beaver Valley Poser Station, has been named to the newly created position of FENOC Chief Operating Officer (COO) and will report to Mr. Saunders. As the new COO, he

will be stationed at Davis-Besse to focus on the restoration and restart of the plant. Reporting to Mr. Meyers will be Guy G.Campbell, Vice President at the Perry Nuclear Power Plant; Howard Bergendahl, Vice President at Davis-Besse, and Mark Bezilla, Vice President at Beaver Valley. His appointment is effective immediately.

L. William Pearce, Plant Manager at Beaver Valley, has been promoted to Vice President - FENOC Oversight and will report to Mr. Saunders. His appointment is effective immediately.

Mark B. Bezilla, currently the Vice President - Technical Support at PSEG, Nuclear LLC, in New Jersey, has been named Vice President at Beaver Valley and will report to Mr. Meyers. His appointment is effective May 20, 2002.

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).

If you deny this NCV, 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 Beaver Valley Power Station.

Requirement Licensee Failed to Meet

Technical Specification 3.3.1.1 requires that a failed RPS channel be returned to an operable status or placed in a tripped condition within 6 hours. Contrary to the above, on November 20, 2001, during the performance of surveillance test 2MSP-36.28-E, control room operators received valid indications that a RCP 4 kV bus underfrequency relay RPS channel had failed. Operators did not recognize that the RPS channel was inoperable and did not place the channel in a trip condition until over 24 hours later. Condition Report 01-7684 was initiated to enter the issue into the CA program. Because redundant channels were operable, this violation is not more than of very low significance, and is being treated as a non-cited violation. **(NCV 50-412/02-04-01)**

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee:

R. Donnellon	Director, Maintenance
L. Freeland	Manager, Nuclear Regulatory Affairs & Corrective Actions
J. Lash	Director, Personnel Development
L. W Pearce	Vice President, FENOC - Oversight & Plant General Manager
M. Pearson	Director, Nuclear Services
F. von Ahn	Director, Plant Engineering

ITEMS OPENED, CLOSED AND DISCUSSED

Closed

50-334/01-05-01	URI	Inadequacy of Hemyc Cable Wrap Fire Barrier Test and Evaluation (Section 1R05.1)
50-412/01-03	LER	Condition Inadvertently Exceeds Technical Specification Allowed Outage Time (Section 1R14.2)
50-334/01-03	LER	Automatic Reactor Trip Due to Low-Low Steam Generator Water Level (Section 4OA3)

Open and Closed

50-412/02-04-01	NCV	Human Error Causes Failure to Place Inoperable Reactor Coolant Pump 4 KV Bus Underfrequency RPS Channel in Trip Within 6 Hours as Required by TS 3.3.1.1. (Section 4OA7)

LIST OF DOCUMENTS REVIEWED

Calculations

Analysis Calculation 8700-DEC-0234 “Fire Wrap Analysis for CH-P1B Power Cable”

Fire Test Report FTCR/94/0125	“Darmatt KM-1 System For Cable Trays”
Fire Test Report FTCR/94/0131	“Darmatt KM-1 System For 3/4" & 4" Steel Conduits”
Fire Test Report FTCR/94/0025	“Darmatt KM-1 System For Intervening Thermal Shorts”
Fire Test Report FTCR/96/0077	“Ampacity Derating Test for 4" and 1" Conduits Encapsulated by KM-1 Darmatt One Hour Replacement Material”
Fire Test Report FTCR/96/0072	“Ampacity Derating factor Test of Cable Tray Encapsulated by KM-1 Darmatt for One Hour Replacement Material
ECP-00025	“Installation and Test Requirements,” Rev. 0

ECP-00025-DIE-1	“50.59 Screen Forms for Fire Wrap Replacement Evaluation,” Rev. 0
ECP-00025-DIE-01	“Feeder Cable Ampacity Evaluation,” Rev. 0
8700-DCM-3072	“Minimum Safety Injection Flows with Uncertainties, Charging Pump Recirculation Flow,” Rev. 3
8700-DCM-3072	“Minimum Safety Injection Flows with Uncertainties, Charging Pump Recirculation Flow,” Rev. 3, Addendum 3.

<u>Drawings</u>

08700-RM-434-4	Instrument Air
10080-RM-436-3	Diesel Starting Air

<u>Procedures</u>

NOP-WM-4001	Foreign Material Exclusion
2OM-7.4A	“Placing a Charging/HHSI Pump in Standby or in Service,” Rev. 18.
2OST-11.16	“Leakage Testing RCS Pressure Isolation Valves,” Rev. 15.

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents Access and Management System
AFW	Auxiliary Feedwater
AOV	Air Operated Valve
CA	Corrective Action
CFR	Code of Federal Regulations
COO	Chief Operating Officer
CR	Condition Report
DBT	Design Basis Threat
DEP	Drill/Exercise Performance
EAL	Emergency Action Level
EOP	Emergency Operating Procedure
EP	Emergency Procedure
EPIP	Emergency Plan Implementing Procedure
EPRI	Electric Power Research Institute
FENOC	FirstEnergy Nuclear Operating Company
HHSI	High Head Safety Injection
IST	Inservice Test
KV	Kilovolt
LEFM	Leading Edge Flow Meter
LER	Licensee Event Report
MR	Maintenance Rule
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NCV	Non-Cited Violation
NUREG	NRC Technical Report Designation
ODCM	Offsite Dose Calculation Manual
OM	Operating Manual
OST	Operational Surveillance Test
PARS	Publicly Available Record
PI	Performance Indicator
PMT	Post-Maintenance Test
psig	Pounds per Square Inch Gauge
RCA	Radiologically Controlled Area
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
REMP	Radiological Environmental Monitoring Program
RPS	Reactor Protection System
RW	River Water
SDP	Significance Determination Process
SG	Steam Generator
SLCR	Supplemental Leak Collection and Release System
SOV	Solenoid Operated Valve
SSC	Structures, Systems, and Components
TLD	Thermoluminescent dosimeter
TM	Temporary Modification
TS	Technical Specification
URI	Unresolved Item