April 27, 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 05000334/2001-002; 05000412/2001-002

Dear Mr. Myers:

On March 31, 2001, the NRC completed an inspection at your Beaver Valley Units 1 & 2. The enclosed report documents the inspection findings which were discussed on April 9, 2001, with Mr. Robert Saunders, yourself, and members of your staff.

This inspection was an examination of 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). 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 actions 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 within 30 days of the date of this inspection report, 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, and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001, and the NRC Resident Inspector at the Beaver Valley facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room and 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/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

Docket Nos.: 05000334; 05000412
License Nos: DPR-66, NPF-73

Enclosure: Inspection Report 05000334/2001-002; 05000412/2001-002

Attachment 1: Supplemental Information
Attachment 2: NRC's Revised Reactor Oversight Process

cc w/encl:
L. W. Pearce, Plant General Manager
R. Fast, Director, Plant Maintenance
F. von Ahn, Director, Plant Engineering
R. Donnellon, Director, Projects and Scheduling
M. Pearson, Director, Plant Services
T. Cosgrove, Manager, Licensing
J. A. Hultz, Manager, Projects and Support Services, FirstEnergy
M. Clancy, Mayor, Shippingport, PA
Commonwealth of Pennsylvania
State of Ohio
State of West Virginia
R. Calvan, Regional Director, FEMA Region III

Distribution w/encl:
Region I Docket Room (with concurrences)
D. Kern, DRP - NRC Resident Inspector
H. Miller, RA
J. Wiggins, DRA
J. Rogge, DRP
N. Perry, DRP
T. Haverkamp, DRP
D. Barss, NRR
J. Shea, OEDO
E. Adensam, NRR
L. Burkhart, PM, NRR
R. Schaff, Backup PM, NRR
P. Milano, Alt PM, NRR

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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos.	05000334, 05000412
License Nos.	DPR-66, NPF-73
Report Nos.	05000334/2001-002, 05000412/2001-002
Licensee:	FirstEnergy Nuclear Operating Company
Facility:	Beaver Valley Power Station, Units 1 and 2
Location:	Post Office Box 4 Shippingport, PA 15077
Dates:	February 11, 2001 through March 31, 2001
Inspectors:	D. Kern, Senior Resident Inspector G. Dentel, Resident Inspector G. Wertz, Resident Inspector J. McFadden, Health Physicist D. Silk, Emergency Preparedness Specialist
Approved by:	J. Rogge, Chief Projects Branch 7 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000334-002, IR 05000412-002, on 2/11-3/31/2001; FirstEnergy Nuclear Operating Company; Beaver Valley Power Station; Units 1 & 2. Fire Protection.

The inspection was conducted by resident inspectors, a regional health physicist, and an emergency preparedness specialist. The inspection identified two Green findings, which were Non-Cited Violations. The significance of the findings are indicated by their color (Green, White, Yellow, Red) using IMC 0609 "Significance Determination Process" (SDP).

A. Inspector Identified Findings

Cornerstone: Initiating Events

- **Green**. The inspectors identified a Non-Cited Violation for failure to properly implement fire protection procedures for control of transient combustible materials including a drum of lube oil and oily rags.

  This finding was of very low safety significance because the detection and suppression fire systems in the areas were unaffected.

- **Green**. The inspectors identified a Non-Cited Violation for failure to establish compensatory measures for an inoperable fire door barrier between the Unit 2 emergency switchgear rooms.

  This finding was of very low safety significance because the likelihood of a fire progressing through the door was considered very low. Therefore, one complete train of safety equipment was likely to remain available to mitigate the accident.

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 100 percent power. On February 23, operators performed a planned power reduction to 45 percent to replace the 'B' main feedwater pump bearing and to investigate main turbine electro-hydraulic control anomalies. The unit was restored to full power on February 25 and remained at or near 100 percent power throughout the inspection period. Operators closely monitored changes in reactor coolant pump (RCP) seal leakoff throughout the period ('A' and 'C' RCP leakoff decreased to approximately 1.1 gallons per minute (gpm) and 'B' RCP leakoff increased to 3.8 gpm.

Unit 2 began this inspection period at 100 percent power. On March 17, the 'A' condensate pump motor tripped on overcurrent, which resulted in an automatic reactor trip (Section 4OA3.1). Operators synchronized the unit to the off-site power grid on March 20, following repairs. The unit achieved full power on March 23 and 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 Control Room Emergency Bottled Air Pressurization System

a. Inspection Scope

The inspectors performed a partial system walkdown of the control room emergency bottled air pressurization system (CREBAPS). The inspectors reviewed the system alignment as shown on drawing 8700-RM-444A-1, Rev. 4, and performed a field verification of the major equipment alignment. The inspectors noted during their walkdowns that 7 of the 20 manual isolation valves located on the 10 pressurized air bottles (2 manual isolation valves per bottle) were smaller than the remaining valves.

The inspectors discussed the apparent discrepancy with the system engineer and valve design engineer who indicated that the smaller valves were original plant equipment which had been purchased to a design specification of 2500 pounds per square inch (psi). (The CREBAPS bottles have a design pressure of 2450 psi.) Additionally, the plant had only five CREBAPS bottles originally but a modification in 1988 installed another five bottles. For the additional five bottles, the valves were not procured to the original pressure rating but to a pressure class (American National Standards Institute [ANSI] B 16.34 pressure class 1500 [3095 psi]). This resulted in 10 substantially larger valves being installed in the system on these 5 bottles. Also, due to a lack of spare parts associated with the original valves, maintenance was performed which replaced 3 of the original 10 valves with the newer style ANSI B 16.34 pressure class 1500 valves leaving the 7 original valves in the system. The inspectors determined that the seven remaining valves purchased to the original design specification of 2500 psi were adequately sized for this installation.

The inspectors reviewed the following design documents in order to verify the authenticity of the CREBAPS manual isolation valves installation and replacement history:

- Condition Report (CR) 99-02382, Various Ball Valves in CREBAPS
- Technical Evaluation Report 8843, Replacement of CREBAPS Ball Valves, Revisions 0, 1, 2, and 3
- Engineering and Design Coordination Report, PS-1019, CREBAPS leakage past valve stem packing, dated November 3, 1975
- Purchase Order BV-19194, Ordering CREBAPS Valves, dated November 12, 1975
- Beaver Valley Manual-123, Tabulation of Stone & Webster Mark Numbers, specification VBS-150Y-CR, 2500 psi ball valve, dated November 30, 1975.
- Engineering Memorandum (EM) 118367, Contromatics ½”, 3/4", and 1" Ball Valve Replacement
- EM-118633, Replacement CREBAPS Ball Valves
- EM-200012, CREBAPS Ball Valve Design Characteristics

b. Issues and Findings

No findings of significance were identified.

.2 Unit 1 High Head Safety Injection System

a. Inspection Scope

The inspectors performed a partial system walkdown of the Unit 1 ‘A’ high head safety injection (HHSI) system. The inspectors reviewed the system alignment to verify it was aligned properly as described in Operating Manual (OM) Figure Number 7-1, Rev. 15, and procedure 1OM-7.3.B.1, "Valve List 1CH," Rev. 16. The Unit 1 ‘A’ HHSI was selected due to the increased risk associated with the 'C' HHSI out-of-service for planned maintenance and the increased probability of misalignment due to painting in the ‘A’ HHSI pump cubicle.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors reviewed the fire protection analyses for both units and identified the following risk significant areas:

- Unit 1 primary auxiliary building 735' elevation (PA-1E)
- Unit 1 cable spreading room (CS-1)
- Unit 2 service building cable tray area (SB-3)
- Unit 2 cable vault and rod control area (CV-3)
- Unit 2 cable tunnel (CT-1)

Specific fire protection conditions examined included control of transient combustibles, material condition of fire protection equipment, and the adequacy of any fire impairments and compensatory measures.

b. Findings

Improper Control of Combustible Material

The inspectors identified a Non-Cited Violation for failure to properly implement fire protection procedures for control of transient combustible materials. This finding was of very low safety significance because the detection and suppression fire systems were unaffected.

During walkdowns of the Unit 1 primary auxiliary building, the inspectors identified the following combustible material control deficiencies:

- On March 7, a 55 gallon drum of lube oil approximately half full of lube oil was not sealed and was left unattended near the Unit 1 HHSI pump area.
- On March 8 and again on March 28, several oily rags were found stored in an open plastic bag near the Unit 1 HHSI pump area.

The inspectors determined that the combustible material control deficiencies were more than a minor issue, because these items, if left uncorrected, could become a more significant safety concern by increasing the frequency and severity of a fire around the Unit 1 HHSI pumps and the reactor component cooling water pumps. The amount of transient combustible materials (lube oil in the 55 gallon drum) in the primary auxiliary building was approximately five times the amount assumed in the fire hazard analysis.

Using the Significance Determination Process of IMC-0609, Appendix F, Fire Protection, the inspectors determined that the controls for transient combustible materials was inadequate and was a degradation of a fire protection feature. The additional transient combustibles did not adversely effect the four fire mitigation defense-in-depth elements listed in Figure 4-1 and therefore, this finding was of very low safety significance (GREEN finding). Further, the significance was mitigated since the drum of lube oil was left unattended and not sealed for only a short time (less than 2 hours) and the fire detection and suppression systems were unaffected in the primary auxiliary building.

Technical Specification (TS) 6.8.1.f specifies that written procedures shall be implemented which cover the Fire Protection Program implementation. Nuclear Power Division Administrative Procedure (NPDAP) 3.5, "Fire Protection," Rev. 14, specifies that: 1) transient combustibles in safety related fire areas shall not be left unattended unless properly stored; 2) all containers of combustible liquids shall be properly closed or sealed when not in use; and 3) oil-soaked rags or oily waste shall be deposited in metallic containers with a lid. Contrary to these requirements, the licensee failed to properly seal and control the lube oil drum in the primary auxiliary building and properly dispose of oily rags. This finding was a violation of TS 6.8.1.f.

This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy **(NCV 05000334/2001-002-01)**. This violation is in the licensee's corrective action program as CRs 01-1264 and 01-1911.

Degraded Fire Barrier

The inspectors identified a Non-Cited Violation for failure to properly implement fire protection procedures for the establishment of compensatory measures for a degraded fire door. This finding was of very low safety significance because one mitigating train of safety equipment was likely to remain available in the event of a fire.

On March 19 during the walkdown of the Unit 2 service building, the inspectors found an emergency switchgear fire door ajar and unlatched. The degraded fire door was previously identified in the work control process on March 11. However, the proper compensatory measures (verify fire detector operability and establish an hourly fire watch patrol) were not established until March 19 when the inspectors informed the nuclear shift supervisor (NSS) of the deficiency. The inspectors found the fire door ajar several times during the next two days. The 8-day delay in establishing an hourly fire watch and repeated failure to verify door closure demonstrated insensitivity to fire barrier integrity.

The inspectors determined that the degraded fire door could have a credible impact on safety, in that, a fire in the ‘A’ emergency switchgear room could potentially progress to the ‘B’ switchgear room. This issue was considered a moderate degradation of the 3-hour barrier between emergency switchgear rooms.

Using the SDP of IMC-0609, Appendix F, Fire Protection, Phase I, the inspectors determined that the loss or degradation of a 3-hour fire barrier was an impairment of a fire protection feature (fire barrier). Using Figure 4-2 of the SDP, and fire protection Scheme 1 to address the emergency switchgear rooms, the inspectors determined that a Phase II analysis was required. The inspectors determined that the degraded fire door was a finding of low safety significance because a credible fire could not be constructed to damage equipment in both emergency switchgear rooms. This determination was based on the configuration of the equipment near the fire door and the observed condition of the degraded fire door. Thus, a train of safety equipment was likely to remain available to mitigate an accident, since a fire in one switchgear room was unlikely to affect equipment in the other train (room). Therefore, the degraded fire barrier was a finding of very low safety significance (GREEN finding).

TS 6.8.1.f specifies that written procedures shall be implemented which cover the Fire Protection Program implementation. NPDAP 3.5, "Fire Protection," Rev 14, specifies that: fire doors separating portions of redundant systems important to safe shutdown within a fire area shall be operable and if not, compensatory measures shall be implemented such as verification of the operability of the fire detectors and establishment of an hourly fire watch patrol within one hour. Contrary to this requirement, the licensee failed to establish the compensatory measures for an inoperable fire door. This finding was a violation of TS 6.8.1.f.

This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy **(NCV 05000412/2001-002-02)**. This violation is in the licensee's corrective action program as CR 01-1916.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report, the Individual Plant Examination, and Individual Plant Examination of External Events to evaluate the design basis and risk significance for internal and external floods. The inspectors also reviewed the TSs, abnormal operating procedure ½ OM-53C.4A.75.2, "Acts of Nature - Flood," Rev. 15, and operating logs to verify procedures and operator actions for coping with floods were appropriate. Based on associated risk significance, the inspectors performed walkdowns of the plant areas listed below. During these walkdowns the inspectors examined a sample of internal and external flood seals, inspected the material condition of potential sources of internal flooding, and verifyied various floor drains, sump pumps, and level alarm circuits were operable. The inspectors compared their inspection results with the most recently completed Beaver Valley Test (BVT), 2BVT-1.33.07, "Flood Seals Visual Inspection," Rev. 1, and discussed observations with the Flood Protection Engineer. CRs or maintenance work requests were written when appropriate to resolve the inspectors' observations. Based on reviewing recently issued CRs, the inspectors determined that station personnel maintained a low threshold for identifying and resolving flood protection issues through the CR program.

- Unit 2 safeguards building (flood zones SG-1N and SG-1S).
- Unit 1 intake structure pump cubicals 'A' and 'D' (flood areas IS-1 and IS-4).
- Unit 1 & 2 external flood protection walkdown including repairs to a failed Unit 2 safeguards building flood seal.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed Unit 1 operator training, focusing on human performance of high-risk operator actions. The inspectors reviewed the operators' ability to correctly evaluate the training scenario and implement the emergency plan. The inspectors also evaluated whether deficiencies were identified and discussed during critiques.

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. For selected systems, the inspectors observed Maintenance Rule Steering Committee (MRSC) meetings to determine whether system performance was properly dispositioned for MR category (a)(1) or (a)(2) performance monitoring.

- On February 20, 23, 28, and March 6 and 7, the Unit 1 ‘A’ inside containment instrument air compressor was removed from service for planned maintenance for 81.5 hours. The maintenance minimally improved the performance of the compressor which had been degrading slowly over the past several months. The containment instrument air system was meeting its maintenance rule goals for (a)(2) status.

- Unit 2 compressed air system availability improved over the past 6 months, and the system engineer recommended the system be restored to category (a)(2) performance monitoring status. The MRSC noted that future scheduled corrective maintenance, which would implement actions intended to improve long term reliability, may result in the system exceeding the established performance criteria. The MRSC directed that the system remain in MR category (a)(1) for continued enhanced performance monitoring.

- Unit 2 reactor coolant system (RCS) leakage resulted in declaration of an Unusual Event and two forced plant shutdowns in December 2000. The MRSC determined that the system exceeded two plant level criteria (unplanned capability loss factor and Emergency Preparedness Plan entries). The committee determined that the two forced shutdowns, each resulted from a maintenance preventable functional failure (MPFF) of packing for an RCS drain valve. The causes of the MPFFs were different and therefore this was not a repetitive MPFF. The committee determined that the station’s Valve Packing Program should be placed in MR category (a)(1) monitoring status due to this being the root cause of the MPFFs.

- A Unit 2 480 volt breaker which supplies power to the ‘B’ train of emergency switchgear ventilation failed on February 4, 2001. The Unit 2 480 volt station service system was already in MR category (a)(1) due to breaker lubrication issues. Engineers determined that the February 4 failure was an MPFF, but unrelated to lubrication issues. The cause was a manufacturing deficiency which resulted in inadequate clearance between the shunt trip actuating device and the auto trip indicator assembly. The failure was an MPFF because the breaker was previously removed from service twice to investigate this problem, and each time was returned to service without identifying or correcting the cause. Extent of condition reviews and recommended corrective actions were appropriate. The system remained in MR category (a)(1) for the lubrication issue. Performance goals did not require revision due to the inadequate clearance issue.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the licensee's scheduling and control of maintenance activities in order to evaluate the effect on plant risk. This review was against criteria contained in Nuclear Power Division Administrative Procedure (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:

- On March 28, the inspectors examined the impact on Unit 1 plant risk of having the 'C' HHSI pump, the 'A' instrument air compressor, and the 1-2 emergency diesel generator out-of-service. In addition, the inspectors reviewed the use of a new computer-based reactor safety risk monitor program.

- The Unit 2 'B' low head safety injection pump was removed from service on February 26 for planned maintenance with an expected unavailability of 10 hours. The work was performed correctly and expeditiously as the pump was returned to an available status in approximately 6 hours.

- The Unit 2 'B' quench spray pump was removed from service on March 29 for planned maintenance with an expected unavailability of 15 hours. The work was performed correctly and expeditiously as the pump was returned to service in approximately 12 hours (see Section R19 for post-maintenance testing [PMT]).

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed operability evaluations in order to determine that proper operability justifications were performed for the following items:

- On March 1, 2001, the Unit 1analog rod position indication (ARPI) for control rod H-10 indicated approximately 12 steps different than the associated group position indication, resulting in repeated control room annunciators. TS 3.1.3.2 requires ARPI to agree within 12 steps of the group step counter. Engineers developed an action plan to verify control rod H-10 position and evaluated H-10 ARPI. The inspectors observed pre-evolution briefings, control rod movement verification tests, and corrective maintenance to adjust the H-10 ARPI temperature null setting. Based on results of the testing, the NSS determined that H-10 ARPI remained operable.

- Several intake structure penetration flood seal issues were identified by engineers during a River Water Latent Issue Review. Additionally, operators identified water leaking through a piping penetration into the Unit 2 safeguards building and inspectors observed leakage past a flood door between the ‘A’ and ‘B’ intake structure pump cubicles. Basis for continued operation 1-01-001 was developed to evaluate the intake structure penetration flood seal issues. CRs and corrective maintenance were initiated to address the other flood seal issues.

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds

a. Inspection Scope

During the performance of Unit 2 routine plant status control room walkdowns, the inspectors noticed numerous equipment deficiency tags on control room equipment. The inspectors reviewed the deficiencies to determine their effect on the operating crews’ ability to monitor and control plant parameters. The inspectors also reviewed Beaver Valley station’s list of control room deficiencies as defined in “Operations Manager Desktop Guide-002,” Rev. 5, to determine if the list accurately reflected control room deficiencies affecting the operators’ ability to control or monitor plant parameters.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed and/or observed several 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.

- The inspectors reviewed the testing and analysis for the new Unit 1 ‘A’ HHSI pump. The inspectors examined the inservice testing data, the brake horsepower data, and other data taken as part of Operations Surveillance Test (OST) 1OST-7.4, "Centrifugal Charging Pump Test [1CH-P-1A]," Rev 17.

- The Unit 2 ‘B’ quench spray pump was post-maintenance tested in accordance with OST 13.2, “Quench Spray Pump [2QSS*P21B] Test,” Rev. 14, following planned preventive maintenance on March 29.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

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

- 1MSP-02.11-I "Power Range Neutron Flux Channel N41 Overpower Quarterly Calibration," Rev. 6

- 1OST-1.1 “Control Rod Assembly Partial Movement Test,” Rev. 4. The inspectors identified that two ARPIs deviated from the group demand step counter by more than 12 steps during the test. Operators did not recognize this degraded condition. CRs 01-1309 and 01-1385 were initiated to address the inspector identified deficiencies regarding application of TS 3.1.3.2 and ARPI inaccuracy.

- 2OST-6.2A “Computer Generated Reactor Coolant System Water Inventory Balance,” Rev. 13

- 1OST-6.2A “Computer Generated Reactor Coolant System Water Inventory Balance,” Rev. 4

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the Units 1 and 2 temporary modifications (TMs) and associated implementing documents to ensure the plant's design basis and effected system or component operabilities were maintained. Nuclear Power Division Administrative Procedure 7.4, "Temporary Modifications," Rev. 9, specified requirements for development and installation of TMs. The inspectors reviewed TMs associated with the following items:

- All Unit 1 TMs for their cumulative impact on safety and operability of safety-related equipment. In addition, the inspectors examined TM 1-01-001, "Installation of RW Bypass Around 1VS-E-3A & B," for the effect on the river water system and the station chiller.

- All Unit 2 TMs for their cumulative impact on safety and operability of safety-related equipment. In addition, the inspectors examined TM 2-96-016, "Installation of a Jumper to Bypass 2CWS-PS117 to enable Cooling Tower Pumps to Start," for the effect on cooling tower pump reliability and effect on the initiating event frequency of a reactor trip and/or loss of normal heat sink.

b. Findings

No findings of significance were identified.

Emergency Preparedness (EP)

EP4 Emergency Plan Reviews

a. Inspection Scope

The inspectors conducted an in-office review of licensee submitted changes for several emergency preparedness documents to determine if the changes decreased the effectiveness of the plan. The review assessed all emergency plan changes and implementing procedures related to the risk significant planning standards in Title 10 of the Code of Federal Regulations (CFR) 50.47(b) (event classification, notification, radiological assessment and protective action recommendations). The reviewed documents are as followed:

- Emergency Plan, Section 3, Rev, 13
- Emergency Plan, Section 5, Rev. 14
- Emergency Plan, Section 6, Rev. 13
- Emergency Plan, Section 7, Rev. 13
- Emergency Plan, Section 8, Rev. 14
- Emergency Plan, Appendix E, Rev. 12

- Emergency Preparedness Plan Implementing Procedure (EPP/IP) 1.1, “Notifications,” Rev. 24
- EPP/IP 1.2, “Communications and Dissemination of Information,” Rev. 15
- EPP/IP 1.4, “Technical Support Center Activation, Operation, and Deactivation,” Rev. 14
- EPP/IP 1.5, “ Emergency Support Center (OSC/ROC) Activation Operation, and Deactivation,” Rev. 11
- EPP/IP 2.6, “Environmental Assessment and Dose Projection Controlling Procedure,” Rev. 12

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed an emergency event training evolution conducted at the Unit 1 control room simulator to evaluate emergency procedure implementation, event classification, and event notification. The event scenario involved safety-related component failures and plant conditions warranting an Alert event declaration. The inspectors observed the critique to determine whether the licensee critically evaluated operator performance to identify deficiencies and weaknesses. Additionally, the inspectors verified the emergency preparedness performance indicators were properly evaluated consistent with Nuclear Energy Institute (NEI) 99-02, “Regulatory Assessment Performance Indicator Guideline,” Rev. 0. Additional documents used for this inspection activity included:

- Emergency Operating Procedure (EOP) E-0, “Reactor Trip or Safety Injection,” Rev. 7
- EOP ES-0.1, “Reactor Trip Response,” Rev. 6
- EOP E-3, “Steam Generator Tube Rupture,” Rev. 6.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope

The inspection included the following activities to determine the effectiveness of access control to radiologically significant areas.

The inspectors observed radioactive material storage areas outside the radiologically controlled area (RCA) and within the protected area, and toured all elevations of the auxiliary and fuel buildings of both units. During these walkdowns, the inspectors reviewed the posting and labeling of radiation and contamination areas and levels, labeling of radioactive material, barricading of contaminated and high radiation areas (HRAs), and the status of locked HRAs. The inspectors also observed activities at the main RCA access control points to verify compliance with requirements for RCA entry and exit, wearing of record dosimetry, and issuance and use of alarming radiation dosimeters (ARDs). The inspectors noted that new exit portal monitors were being used to initiate a passive internal monitoring program.

The inspection included a review of the following procedures, records, and documents.

- Nuclear Operating Administrative Procedure NOP-LP-2001, “CR Process,” Rev. 1
- Health Physics Procedure RP 8.1, “Radiological Work Permit,” Rev. 14
- Radiological Engineering Administrative Procedure 1.103, “Control of Engineering Calculations,” Rev. 6
- Radiation Work Permit (RWP) 101-1036, Overhaul charging pump/repair discharge head gasket/replace pump seals and/or pump/ and replace oil piping
- RWP 301-3002, Expose Thermoluminescent Dosimeters (TLDs)/ ARDs/ Dosimeters
- Certificate for the vendor-supplied TLD service used for the whole-body badge of record (National Voluntary Laboratory Accreditation Program of the National Institute of Standards and Technology)
- BVPS Neutron Energy Spectrum and Dosimetry Study, June 1992
- Calculation ERS-AJL-98-003, “Passive Internal Monitoring Program,” Rev. 2
- Health Physics Section, Year 2001 Business Plan

The inspection reviewed eight (8) CRs that addressed worker and/or radiation protection technician performance errors or radiological protection concerns (CR Nos. 00-4326, 01-0314, 01-0498, 01-0524, 01-0532, 01-0594, 01-0661, and 01-0850), occurring between December 12, 2000 and February 16, 2001. The review included an evaluation of the associated cause evaluations and corrective actions.

The review was against criteria contained in 10 CFR Parts 20.1201 (Occupational dose limits for adults), 20.1204 (Determination of internal exposure), 20.1208 (Dose equivalent to an embryo/fetus), Subpart F (Surveys and monitoring), 20.1601 (Control of access to high radiation areas), Subpart H (Respiratory protection and controls to restrict internal exposures in restricted areas), and 20.1902 (Posting requirements) and against TS 6.12 (High Radiation Area) and site procedures (cited above in this section).

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Control

a. Inspection Scope

The inspection included the following activities to determine the effectiveness of ALARA (As Low As Reasonably Achievable) planning and control.

The inspectors reviewed the following procedure, records, and documents.

- NPDAP 3.1, “Exposure Control,” Rev. 5
- ALARA Review No. 01-1-02 for RWP 101-1036
- ALARA report for Unit 2's eighth refueling outage (September 23 to October 25, 2000)
- Nuclear ALARA Review Committee Meeting Minutes for Meeting on February 7, 2001
- Draft Beaver Valley Power Station (BVPS) 2001 ALARA Initiatives

The inspectors reviewed the lessons learned from the post-outage ALARA report for the Unit 2 refueling outage, the collective exposure results for the calendar year of 2000, and the site person-rem goal for the calendar year of 2001. The inspectors noted that the Nuclear ALARA Review Committee had met formally at the frequency required by NPDAP 3.1 during the calendar year of 2000.

The review was against criteria contained in 10 CFR 20.1101 (Radiation protection programs) and 20.1701(Use of process or other engineering controls) and in site procedures (cited above in this section).

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

.1 Reactor Coolant System Specific Activity

a. Inspection Scope

The inspectors reviewed the Unit 1 and Unit 2 performance indicator (PI) for RCS specific activity for the period March 2000 through February 2001. 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 RCS specific activity data reported was consistent with NRC approved guidance, provided in NEI 99-02, "Regulatory Assessment PI guideline," Rev. 0. In addition, the inspectors reviewed the CRs 00-3166 and 00-2611 associated with this PI.

b. Findings

The inspectors identified that the Units 1 and 2 chemists take additional samples beyond the TS surveillance required sampling. These additional samples were taken using a different method, but similar to the TS methodology. This additional data, RCS strip isotopic data, was not included in determining the maximum monthly dose equivalent iodine for the PI. CR 01-1763 was generated to evaluate whether this data should have been included. The engineer assigned to this PI planned to submit a frequently asked question to the NRC on this issue. The significance of this issue was minor, in that, the indicator would only increase slightly and would remain below the White threshold.

.2 Reactor Coolant System Identified Leak Rate

a. Inspection Scope

The inspectors reviewed the Unit 1 and Unit 2 PI for unidentified RCS leak rate for the period January through December 2000. The accuracy of reported data was verified by reviewing selected monthly operating reports, shift operating logs, Licensee Event Reports (LERs), and surveillance tests. Unit 2 experienced elevated RCS leakage for short time periods in July and December 2000, due to a leaking charging pump casing drain valve and a packing leak from a RCS drain valve respectively. The inspectors reviewed detailed records for the July and December 2000 periods to determine whether the RCS leak rate data reported was consistent with NRC approved guidance, provided in NEI 99-02, "Regulatory Assessment PI guideline," Rev. 0. Additionally, the inspectors observed operators perform the TS required periodic RCS leak rate surveillance test on each unit.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

.1 Unit 2 Automatic Reactor Trip due to Low Steam Generator Level

a. Inspection Scope

On March 17, 2001, at 9:12 p.m., Unit 2 automatically tripped from 100 percent power. One of two running condensate pumps tripped (the third condensate pump was out of service for motor overhaul), causing a loss of suction to the main feedwater pumps, and a subsequent automatic reactor trip due to low-low steam generator (SG) water level (CR 01-1488). 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 discussed the plant status with operators to verify safe plant conditions. Surveillance tests on SG atmospheric steam dumps were in progress at the time of the trip. The inspectors determined that these surveillances did not cause the event. The inspectors also verified the reactor trip was properly reported in accordance with 10 CFR 50.72. 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 (approximately 2E-6) and that no additional NRC reactive response was necessary. The apparent cause of the condensate motor trip was excessive phase 'C' current due to galvanic corrosion of dissimilar metals at the electrical connector termination.

b. Findings

No findings of significance were identified.

.2 (Closed) LER 05000334/2000-07: Technical Specification Non-Compliance Due to Misinterpretation of Containment Isolation Valve Requirements for GDC 57 Penetrations. This event was discussed in NRC Inspection Report Nos. 50-334(412)/2000-010. No new issues were revealed by the LER. This LER was closed during an onsite review.

.3 (Closed) LER 05000412/2001-01: ESF Actuation of Feedwater Isolation While Shutting the Plant Down for Refueling.

a. Inspection Scope

The inspectors reviewed the licensee event report, safety significance, causal assessment and corrective actions. The inspectors determined that the safety significance of the event was very minor. The event was accurately captured in the corrective action program as CR 00-3203 and the completed corrective actions were appropriate. However, the inspectors determined that the LER causal assessment was incomplete because as it did not recognize the contribution of procedural compliance to the event. (See Section 4OA7). The inspectors determined that the cause of the event was due to the operating crew not maintaining the specified SG level. The inspectors discussed the apparent discrepancy with the Unit 2 Operations manager who initiated CR 2001-1860. This LER was closed during an onsite review.

b. Findings

No findings of significance were identified.

4OA5 Administrative Review to Previous United States Nuclear Regulatory Commission Inspection Report Closed Items

Nuclear Regulatory Commission Inspection Report 05000334/2001-003 contained incorrect numbers for the closed violations. The items incorrectly closed in the report are EA-00-045, Items 2013 and 2014. The correct items closed are EA-00-053, Items 02013, 03013, and 3023. The closed items are also listed in the list of items closed at the end of this report.

4OA6 Management Meetings

.1 Exit Meeting Summary

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 April 9, 2001. 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 March 20-21, 2001, Mr. John Rogge, Chief, Reactor Projects Branch 7, toured Beaver Valley Power Station and met with station personnel to review plant performance.

.3 Management Drop-In Visit

On March 30, 2001, Mr. Lew Myers, Senior Vice President, FENOC and Mr. Fred von Ahn, Director, Engineering, Beaver Valley Power Station met with Mr. Hubert Miller, NRC Region I Administrator and other NRC staff in King of Prussia, Pennsylvania. Topics of discussion included the December 2000 Unit 2 reactor coolant system leak event, current plant issues, and ongoing initiatives to identify and address latent equipment issues.

4OA7 Licensee Identified Violation

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

NCV Tracking Number,

NCV 05000412/2001-002-03

Requirement Licensee Failed to Meet

TS 6.8.1 requires written procedures covering the activities recommended in Appendix 'A' of Regulatory Guide 1.33, Revision 2, February 1978. Station Operating Procedure 2OM-51.4.L, "Station Shutdown from 40 percent Power to Mode 5," Rev. 0, specified SG level be maintained within the alarm setpoint control bands. Contrary to the above, operators maintained SG levels approximately 10 percent above the alarm setpoint control bands during plant shutdown on September 23, 2000. A subsequent inappropriate rate of increase of steam dump pressure resulted in a steam generator swell and automatic feedwater isolation due to high SG level. Reference CR 00-3203. This is being treated as a Non Cited-Violation.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

a. Key Points of Contact

T. Cosgrove, Manager, Licensing
R. Donnellon, Director, Projects and Scheduling
J. Duvall, Radiation Control Technician
R. Fast, Director, Plant Maintenance
J. Fontaine, Health Physics Specialist
R. Freund, Rad Ops Supervisor, Unit 2
M. Helms, Senior Health Physics Specialist
J. Lebda, Supervisor-Radiological Engineering and Health
L. Myers, Senior Vice President, FENOC
M. Pearson, Director, Plant Services
R. Pucci, Health Physics Specialist
R. Saunders, Chief Nuclear Officer, FENOC
B. Sepelak, Licensing Specialist
J. Sipp, Health Physics Manager
F. von Ahn, Director, Plant Engineering
L. W. Pearce, Plant General Manager
D. Weitz, Senior Health Physics Specialist

L. Ryan, Division of Nuclear Safety Engineer, PA Dept of Environmental Protection

b. List of Items Opened, Closed and Discussed

Opened/Closed

05000334/2001-002-01	NCV	Failure to Properly Control Combustible Materials in Accordance with the Fire Protection Program (Section 1R05)
05000412/2001-002-02	NCV	Failure to Establish the Compensatory Measures for an Inoperable Fire Door in the Unit 2 Emergency Switchgear Room. (Section 1R05)
05000412/2001-002-03	NCV	Failure to Follow Plant Shutdown Procedure Resulted in Automatic Feedwater Isolation (Section 4OA7)

Closed

05000334/2000-07	LER	Technical Specification Non-Compliance Due to Misinterpretation of Containment Isolation Valve Requirements for GDC 57 Penetrations (Section 4OA3.2)
05000412/2001-001	LER	ESF Actuation of Feedwater Isolation While Shutting the Plant Down for Refueling (Section 4OA3.3)
05000334/EA-00-053, Item 0213	EA	Inadequate Temporary Modification of River Water Pump Seal Water System, (Closed in NRC Inspection Report 05000334/2001-003)
05000334/EA-00-053, Item 3013	EA	River Water Pump Seal Water Supply Design Deficiencies (Closed in NRC Inspection Report 05000334/2001-003)
05000334/EA-00-053, Item 3023	EA	River Water Pump Surveillance Testing Deficiencies (Closed in NRC Inspection Report 05000334/2001-003)

c. List of Acronyms

ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
ARD	Alarming Radiation Dosimeter
ARPI	Analog Rod Position Indication
BVPS	Beaver Valley Power Station
BVT	Beaver Valley Test
CFR	Code of Federal Regulations
CR	Condition Report
CREBAPS	Control Room Emergency Bottled Air Pressurization System
EM	Engineering Memorandum
EOP	Emergency Operating Procedure
EP	Emergency Preparedness
FENOC	FirstEnergy Nuclear Operating Company
gpm	gallons per minute
HHSI	High Head Safety Injection
HRA	High Radiation Area
LER	Licencee Event Report
MPFF	Maintenance Preventable Functional Failure
MR	Maintenance Rule
MRSC	Maintenance Rule Steering Committee
MSP	Maintenance Surveillance Procedure
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NPDAP	Nuclear Power Division Administrative Procedure
NRC	Nuclear Regulatory Commission
NSS	Nuclear Shift Supervisor
NUREG	NRC Technical Report Designation
OM	Operating Manual
OST	Operations Surveillance Test
PI	Performance Indicator
PMT	Post-Maintenance Testing
psi	pounds per square inch
RCA	Radiologically Controlled Area
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RWP	Radiation Work Permit
SG	Steam Generator
SPD	Significance Determination Process
SSC	Structures, Systems and Components
TLD	Thermoluminescent Dosimeter
TM	Temporary Modification
TS	Technical Specification

ATTACHMENT 2

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.