January 26, 2001

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

SUBJECT: BEAVER VALLEY POWER STATION - NRC INSPECTION REPORT 05000334/2000-012; 05000412/2000-012

Dear Mr. Myers:

On December 30, 2000, the NRC completed an inspection at your Beaver Valley Units 1 & 2. The enclosed report documents the inspection findings which were discussed you and other members of your staff on January 5, 2001.

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 one issue of very low safety significance (Green). The issue involved ineffective corrective actions to resolve leakage from a reactor coolant system valve and was determined to be a violation of NRC requirements. However, because of the low safety significance and because the issue was entered into your corrective action program, the NRC is treating the issue as a Non-cited violation, in accordance with Section VI.A of the NRC's Enforcement Policy. If you deny the Non-cited violation, 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, D.C. 20555-0001, with copies to the Regional Administrator, Region I, and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 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 7
Division of Reactor Projects

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

Enclosure: Inspection Report 05000334/2000-012; 05000412/2000-012

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

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
C. O'Daniell, DRP
J. Shea, OEDO
E. Adensam, PD1, NRR
M. Gamberoni, NRR
L. Burkhart, NRR
R. Schaaf, NRR

ADAMS ACCESSION NUMBER: ML010260370

ADAMS DOCUMENT TITLE: IR 05000334-012, IR 05000412-012, on 11/12-12/30/2000; FirstEnergy Nuclear Operating Company; Beaver Valley Power Station; Units 1 & 2. Event Follow-up.

☒ Publicly Available ☐ Non-Publicly Available ☐ Sensitive ☒ Non-Sensitive

OFFICE	RI/DRP	RI/DRP		
NAME	DKern	JRogge		
DATE	01/26/2001	01/26/2001		

OFFICIAL RECORD COPY

U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos. License Nos.	05000334, 05000412 DPR-66, NPF-73
Report Nos.	05000334/2000-012, 05000412/2000-012
Licensee:	FirstEnergy Nuclear Operating Company
Facility:	Beaver Valley Power Station, Units 1 and 2
Location:	Post Office Box 4 Shippingport, PA 15077
Dates:	November 12, 2000 through December 30, 2000
Inspectors:	D. Kern, Senior Resident Inspector G. Dentel, Resident Inspector G. Wertz, Resident Inspector J. McFadden, Health Physicist T. Fish, Operations Engineer J. Caruso, Operations Engineer
Approved by:	J. Rogge, Chief Projects Branch 7 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000334-012, IR 05000412-012, on 11/12-12/30/2000; FirstEnergy Nuclear Operating Company; Beaver Valley Power Station; Units 1 & 2. Event Follow-up.

The inspection was conducted by resident inspectors, two regional operator licensing examiners, and a regional health physics inspector. The inspection identified one Green finding which was a Non-cited violation. The significance of issues is indicated by their color (Green, White, Yellow, Red) using IMC 0609 "Significance Determination Process" (SDP) (See Attachment 1).

A. Inspector Identified Findings

Cornerstone: Barrier Integrity

- **Green.** The inspectors identified a Non-Cited Violation for inadequate corrective actions during packing gland eyebolt replacement for four valves (including 2RCS-557B, the valve that was the cause of the Unit 2 Unusual Event for excessive reactor coolant system (RCS) leakage on December 11, 2000). Mechanics failed to properly consolidate valve packing following corrective maintenance and unknowingly damaged valve packing on all four valves. Station personnel did not fully understand the effects of their corrective maintenance, the design of the valve packing configuration, or properly address the cause for post-maintenance packing leakage prior to restarting the unit on December 15. Consequently, 2RCS-557B valve packing failed and initiated a 5 gallons per minute identified RCS coolant leak. The event revealed knowledge, work instruction, and work practice deficiencies associated with the station's valve packing program implementation.

  The finding was of very low safety significance because the event did not create an open pathway in the physical integrity of the reactor containment barrier or adversely affect the ability to control containment pressure. Failure to adequately perform corrective maintenance to resolve excessive packing leakage from 2RCS-557B following the December 11, 2000, Unusual Event was a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI "Corrective Actions," consistent with Section VI.A of the Enforcement Policy, issued May 1, 2000 (65 FR 25368). (Section 4OA3.1)

b. Licensee-Identified Findings

A violation of very low significance which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee were reasonable. 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 November 29, 2000, operators performed an unplanned power reduction (e.g., <72 hours advance notice) to 65 percent power in preparation for a Technical Specification (TS) required shutdown due to an inoperable turbine driven auxiliary feedwater (AFW) pump (see Section 1R13). The pump was successfully repaired and the unit returned to full power on November 30. Power was briefly reduced to 95 percent on December 26 due to degraded main condenser vacuum while performing main condenser waterbox cleaning.

Unit 2 began this inspection period at 100 percent power. On December 11, excessive unidentified reactor coolant system (RCS) leakage required a plant shutdown and declaration of an Unusual Event. The leakage was subsequently determined to be from a reactor coolant loop drain valve. The unit was restarted (to 8 percent power) and shut down on December 15, due to RCS leakage from the same loop drain valve (see Section 4OA3.1). On December 19, operators restarted the unit after successfully correcting the RCS leakage condition and achieved full power.

1. REACTOR SAFETY
Cornerstone: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors reviewed the station's cold weather protection adequacy in accordance with the following operating surveillance tests (OST) and preventive maintenance procedures (PMP):

- 1OST-45.11 — Cold Weather Protection Verification, Rev. 11
- 2OST-45.11 — Cold Weather Protection Verification, Rev. 13
- 1-PMP-E-45-401 — Heat Trace Circuitry Operability and Setpoint Check for Freeze Protection Circuits, Rev. 1.
- 2PMP-45-HEAT-TRACE-1E — Heat Trace Circuitry Operability Checks, Rev. 7

The inspectors reviewed the outstanding work deficiencies noted in the cold weather protection OSTs and verified that they were of minor significance and properly captured in the corrective maintenance program. The preventive maintenance procedures were reviewed to verify that the calibrations were performed correctly. The inspectors performed a walkdown of the Unit 2 safety-related heat tracing control panels and heat trace for the refueling water storage tank exposed piping that supplies the safety-related quench spray and low head safety injection systems. The inspectors reviewed the heat trace alarm response procedures and interviewed control room operators to assess their understanding of cold weather protection for equipment and associated alarms.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments

a. Inspection Scope

The inspectors performed partial system walkdowns of the Unit 1 high head safety injection and river water systems. The inspectors reviewed the system alignment as shown on plant drawings 8700-RM-407-1, Rev 15 and 8700-RM-430-1, Rev 16 and performed field verification of major equipment alignment.

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 cable spreading room (Area CS-1)
- Unit 2 control building - main control room (Area CB-3)
- Unit 2 cable vault and rod control area (Area CV-1)
- Unit 2 cable vault and rod control area (Area CV-2)

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. Selected safety train cable separation verifications were performed.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

.1 Unit 2 Biennial Operator Requalification Review

a. Inspection Scope

Inspectors reviewed Unit 2 operating history based on assessments from various NRC inspection reports, licensee condition reports, and the NRC plant issues matrix. The senior resident inspector was consulted for insights regarding plant operating history. The inspectors reviewed events that indicated deficiencies in licensed operator performance (for example, Unit 2 experienced a feedwater isolation during a plant shutdown) and verified that facility training staff had addressed performance deficiencies through appropriate training methods, such as classroom lectures and/or simulator exercises. Inspectors reviewed a sample of written exams from the September 1999 biennial exam and the annual operating exams for November 2000. Inspectors noted exam content and quality met the requirements of 10 Code of Federal Regulations

(CFR) 55.59 and guidance of NUREG-1021, “Operator Licensing Examination Standards for Power Reactors.”

Operating exams administered to different crews were compared and the inspectors noted minimal duplication. The inspectors observed the training staff administer the operating test to one shift crew and one administrative crew, and also observed the facility’s evaluation of crew and individual operator performance. Inspectors reviewed the training staff’s response to student feedback on training, and also reviewed remedial training records of operators who had failed some portion of the exam. Lastly, inspectors reviewed a sample of medical records, training attendance records, and license reactivation records.

b. Findings

No findings of significance were identified.

.2 Unit 2 Quarterly Operator Requalification Review

a. Inspection Scope

The inspectors observed Unit 2 operator training, focusing on human performance of time critical tasks. 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 issue 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 following issue was evaluated:

- On November 28, the Unit 1 turbine driven AFW pump experienced a functional failure during post-maintenance testing. The failure and subsequent extent of condition inspections of two additional AFW pumps, increased AFW safety system unavailability by approximately 75 hours. Upon disassembly, mechanics identified that a balance drum capscrew had broken and caused the rotating assembly to mechanically bind. Engineers subsequently determined that general corrosion and residual manufacturing stresses caused the localized bolt failure. Engineers further determined that the bolting material was different from what the vendor had certified and that vendor specified capscrew torque values were incorrect. Although these two issues did not cause the failure, engineers initiated a 10 CFR 50 Part 21 review for generic industry applicability. Corrective actions included disassembly and inspection of the two motor driven AFW pumps.

b. Findings

No findings of significance were identified.

R13 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. The inspectors reviewed the routine planned maintenance and emergent work for the following equipment removed from service:

- The Unit 1 turbine driven AFW pump (1FW-P-2) was taken out of service for planned maintenance with an expected duration of 36 hours. A pump failure during post-maintenance testing extended the planned outage beyond the end of the TS allowed outage time and placed the plant in a 6 hour TS shutdown action statement. Planned maintenance on other risk significant components, including realignment and testing of high head safety injection pumps, was postponed due the elevated plant risk profile of the extended AFW pump outage. Maintenance, engineering, and operations personnel worked continuously and successfully restored the turbine driven AFW pump in time to avoid a TS required shutdown.

- The Unit 1 motor driven AFW pumps (1FW-P-3A and 1FW-P-3B) were disassembled to inspect and replace the balance drum capscrews. This was performed as extent of condition corrective actions for the failed turbine driven AFW pump discussed in Section 1R12 above. Maintenance, engineering, and operations department personnel coordinated effectively to plan, implement, and post-maintenance test the capscrew replacement. The corrective maintenance on 1FW-P-3A required only 16 hours of the 72 hour allowed outage time.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions

a. Inspection Scope

The inspectors reviewed operator performance during the following nonroutine plant evolutions:

- Unit 2 control room operators responded to an unexpected RCS pressure control system problem on November 22, when annunciators A4-1E, “Pressurizer Control Press, Deviation High/Low,” and A4-2G, “Backup Heater Group Auto-On/Off,” alarmed. The operators correctly followed their alarm response procedures and took manual control of RCS pressure. That afternoon, when maintenance technicians removed the suspected unreliable process control card for calibration, the pressure controller, operating in manual, unexpectedly failed downscale and additional pressurizer heaters energized. The operator correctly responded and instructed the technicians to restore the process card. The unexpected pressure control response was due to poor procedure quality. Maintenance procedure 2MSP-6.48-I, “Pressurizer (2RCS*PRE21) Pressure Control Loop 2RCS-P444 Calibration,” Rev. 6, did not clearly describe operator actions to establish manual pressure control prior to removing the process control card. The inspectors reviewed the event with the control room operators and maintenance supervisor. Condition Report (CR) 00-4113 documented the initial pressure control problem and CR 00-4412 documented the procedure deficiency which challenged the operator.

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. In addition, where a component was determined to be inoperable, the inspectors verified the TS limiting condition for operation implications were properly addressed.

- Unit 2 RCS Loop I and II Over Temperature Delta Temperature (OT ΔT) and Over Pressure Delta Temperature (OPΔT) lead/lag signal processing cards (2RCS-TX412P and 2RCS-TX422P) were found to have non zero input potentiometer “Course” and “Fine” settings. These input potentiometer settings affect the lag function of the OTΔT and OPΔT process signals and should be set such that there is no lag on the process signal as described in TS table 2.2-1, notes 1 and 3 (τ_3 = 0). The inspectors reviewed the impact on operability of the two protection channels with the non zero input settings.

  Engineers performed analytical calculations of the effect of the as-found settings on the lead/lag function and determined that the settings would result in a process signal output leading its input and not lagging. Therefore, the OTΔT and OPΔT protective functions remained operable. Additionally, the inspectors also reviewed the OTΔT and OPΔT instrument response time tests which were performed satisfactorily during the last refueling outage. Station management initiated several CRs on this issue (CR 00-3956, 3962, 3941, and 3978). See Section 1R17 for related information.

- On November 22, while performing a quarterly TS surveillance test, technicians identified that the 2-1 station battery average specific gravity (SG) of 1.203 was below the TS Category B limit (>1.205). Operators applied the appropriate TS action statement, which required battery SG restoration above 1.205 within 7 days or a plant shutdown within the following 6 hours. Battery average SG remained below the TS Category B limit following a 100-hour equalizing charge. System engineers determined that electrolyte stratification from a discharge test performed approximately 6 weeks earlier had occurred. Engineers revised the surveillance procedure in order to obtain the average SG readings from the top, middle, and bottom of each battery cell. Using this method, the measured average SG was 1.211. The inspectors reviewed past surveillance test procedures, the vendor technical manual, and consulted NRC battery specialists to verify this method of resolution was technically sound. Engineers initiated CRs 00-4125 and 00-4175 to document this issue.

- While preparing for a planned power uprate, the nuclear safety system supplier (NSSS) vendor identified that the Beaver Valley Unit 1 & 2 TS for inoperable main steam safety valves (MSSV) did not contain a requirement to reduce the power range high neutron flux trip setpoints. The NSSS vendor concluded that when more than one MSSV is inoperable, reduced trip setpoints may be necessary to protect a steam generator from overpressurization during a reactivity insertion accident. The licensee developed basis for continued operation (BCO) evaluations for each unit and implemented compensatory measures pending submittal and approval of a TS amendment request to address the reduced setpoint issue. The inspectors reviewed BCO 2-00-003, “Unit 2 Reduced Trip Setpoints for Inoperable MSSVs” and CR 00-4426. The inspectors confirmed that there has not been more than one inoperable MSSV on a steam generator for either unit since June 1999, when the reduced trip setpoint was removed from TS 3.7.1.

- A failed balance drum capscrew caused the Unit 1 turbine driven AFW pump to become inoperable (see section 1R12). Station personnel developed BCO 1-00-007, “Operability Determination of 1FW-P-3A, 3B,” to evaluate operability of the two motor driven AFW pumps pending inspection of their capscrews. The inspectors concluded the BCO provided reasonable assurance of pump operability.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed Technical Evaluation Report (TER) No. 11823, "NSSS Control & Protection System Time Constant Evaluation," related to OTΔT and OPΔT time constants as a result of the calibration issues described in section 1R15. The inspectors determined that the recommended changes to maintenance procedures 2MSP-6.38(39)(40)-I, "Reactor Coolant Temperature Loop 2RCS-T412(422)(432) delta T-Tavg Protection Channel I(II)(III) Calibration," Rev. 10(11)(10), specified by the TER were incomplete. In addition, some of the recommended procedure changes were not performed. As a result of these problems, engineers performed an extensive review of all other procedure changes specified by TER 11823. An extent of condition review included eight other TER's associated with the reactor protection system. A few other minor procedure discrepancies resulted from this review but no issue challenged reactor protection system operability. Engineers documented this issue in CR 00-3975 and performed an extensive root-cause investigation.

.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed and/or observed several post-maintenance tests (PMTs) to ensure: 1) the PMT was appropriate for the scope of the maintenance work completed; 2) the acceptance criteria were clear and demonstrated operability of the component, and 3) the PMT was performed in accordance with procedures. The following PMTs were observed:

- 2MSP-E-39-001, "Vital Bus Batteries, Test and Inspection," Rev. 5, for restoration of 2-1 station battery following a 100 hour equalize charge. See Section 1R15 for more information regarding station battery 2-1.

- 2OST-13.1, "Quench Spray Pump [2QSS*P21A] Test," Rev. 14, for restoration of the 'A' quench spray pump following preventive maintenance activities.

- 1OST-24.4, "Steam Turbine Driven Auxiliary Feed Pump Test [1FW-P-2]," Rev. 15 was performed following repair of the Unit 1 turbine driven AFW pump balance drum (see Sections 1R12 and 1R13).

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed and reviewed the following operational surveillance tests (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.

- 2OST-1.11B "Safeguards Protection System Train 'A' SIS GO Test," Rev. 20.
- 1MSP-1.04-I "Solid State Protection System Train 'A' Bi-Monthly Test," Rev. 12.

b. Findings

No issues 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, event notification, and protective action recommendation development. 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 Drill/Exercise Performance (DEP) Indicators. The inspectors observed the drill critique to determine whether the licensee critically evaluated drill performance to identify deficiencies and weaknesses. The critique included an additional protective action recommendation training evolution. Additionally, the inspectors verified the DEP performance indicators were properly evaluated consistent with Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guideline," Rev. 0. The inspectors observed that the definition of "Timely," used in station procedures for DEP performance indicator evaluation, differed from NEI 99-02. Procedure revisions were being developed and training evaluators were being briefed to address the discrepancy. 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-2, "Faulted Steam Generator Isolation," Rev. 4
EOP E-3, "Steam Generator Tube Rupture," Rev. 6
Emergency Plan Implementing Procedure (EPIP) IP 1.1, "Notifications," Rev. 23
EPIP I-1a, "Recognition and Classification of Emergency Conditions," Rev. 6
Emergency Preparedness -16, "NRC EPP Performance Indicator Instructions," Rev. 0

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

The inspection included the following activities to determine the accuracy and operability of radiation monitoring instruments that are used for the protection of occupational workers and to determine the adequacy of the program to provide a self-contained breathing apparatus (SCBA) to occupational workers.

In order to evaluate the familiarity of workers with the operation and use of SCBAs, the inspectors observed four individuals (two Radiation Protection Technicians, an Instrumentation and Control Supervisor, and a Mechanical Maintenance Supervisor) as they prepared and donned closed-circuit SCBAs and entered the Unit 2 containment to investigate the cause of a RCS leak. The inspectors walked down various areas in the plant to verify the operability of radiation monitoring equipment and instrumentation, including various elevations of the Unit 1 and Unit 2 primary auxiliary buildings and the Unit 2 fuel handling building. The inspectors reviewed the operability of various installed radiation monitors and portable health physics instrumentation. The inspectors toured the areas in Unit 1 and Unit 2 where closed-circuit and open-circuit SCBAs were stored. The inspectors examined the areas in the Emergency Response Facility where the whole body counters, the personnel ThermoLuminescent dosimeter (TLD) processing equipment, the calibration equipment for the TLDs and personnel electronic dosimeters, and additional SCBA cleaning, sanitizing, repairing, and fit-testing facilities and equipment were located.

The inspection included a review of the most recent calibration records for the following installed radiation monitors to verify adequate calibration status.

- Unit 1 containment high range area radiation monitors 219 A and B
- Unit 1 fuel building ventilation exhaust radiation monitors 103 A and B
- Unit 1 fuel pool bridge area radiation monitor 207
- Unit 2 containment high range area radiation monitors 206 and 207
- Unit 2 control room area radiation monitor 201
- Unit 2 containment airborne radiation monitors 303 A and B

The inspectors examined the procedures and selected recent calibration records for the following types of health physics instrumentation to verify their calibration status: RO-2s, RO-2As, teletectors, hand-held friskers, neutron survey meters, personnel contamination monitors (PCM-1Bs and PCM-2s), portal monitors (PM-6s), and the whole-body counter (fast scanning model).

In addition, the inspectors reviewed the following procedures and documents to evaluate their adequacy:

- Health Physics Program Audit No. BV-C-99-14
- Activity versus time (December 2000) measurement evaluation for the Unit 2 reactor building containment airborne (gaseous) radiation monitor (2RMR-RQ303B)
- Radiological instrument procedure (RIP) 1.2, “Radiation Monitoring System Area Monitor Calibrator (model 848-8 field calibration kit),” Rev. 2
- RIP 1.9, “Model 89-400 Gamma Calibration System,” Rev. 3
- Calibration source certificates for traceability to National Institute of Standards and Technology for sources used to calibrate the whole-body counter and the Unit 1 radiation monitors 103 A and B
- Technical position for passive internal monitoring program
- Health Physics Manual, Appendix 6, ”Respiratory Protection Program,” Rev. 5
- “Respiratory Device/Training/Job Position Requirements Guidelines, Training Administrative Manual,” Rev. 7
- Lesson plan, “Respiratory Protection, MSA-401 SCBA Operation and Use, LP-RP-09,” Rev. 5
- Respirator Fit Program Device issue report
- Radiological procedure (RP) 10.22, “Emergency SCBA Weekly Surveillance,” Rev. 1
- Most recent weekly records for SCBA inspection, air cylinder service life expiration, and air cylinder hydro test expiration
- Revised Respirator Facial Hair Policy dated June 11, 1998
- RP 10.13, “MSA Self-Contained Breathing Apparatus,” Rev. 13
- RP 10.24, “Maintenance of the BioPak 240P Respirator,” Rev. 4

The inspection reviewed five CRs that addressed worker and/or radiation protection technician performance errors or radiological protection concerns (00-1387, 00-1528, 00-1705, 00-3763, and 00-4327), occurring between March 29 and December 13, 2000. The review included an evaluation of the associated cause evaluations and corrective actions.

The review in this part (2OS3) was against criteria contained in Title 10 of CFR Parts 20.1203 (Determination of external dose from airborne radioactive material), 20.1204 (Determination of internal exposure), Subpart F (Surveys and monitoring), Subpart H (Respiratory protection and controls to restrict internal exposures in restricted areas), Subpart L (Records), and against criteria contained in site procedures (cited above in this section).

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA3 Event Follow-up

.1 Unit 2 Unusual Event due to Excessive Unidentified Reactor Coolant System Leakage

a. Inspection Scope

On December 11, at 3:20 a.m., Unit 2 control room operators received alarms indicating a potential leak in the reactor coolant system (RCS) pressure boundary. Indications included a containment gas alert alarm, decreasing pressurizer level, and increasing charging flow. Operators determined that the leak exceeded the one gallon per minute (gpm) TS limit for RCS unidentified leakage and commenced a plant shutdown in accordance with 2OM-53C.4.2.51.1, “Emergency Shutdown,” Rev. 7. At 5:36 a.m., operators estimated the leakrate to be 12-20 gpm. The Nuclear Shift Supervisor (NSS) declared an Unusual Event (UE), based on RCS unidentified leakage exceeding 10 gpm. A UE is the least severe of the four emergency event classifications described in the station’s emergency plan. The technical support center (TSC) was staffed but not activated. The inspectors responded to the control room and the TSC to evaluate plant equipment, mitigating system availability, operator actions including communications and use of emergency operating procedures, and plant stabilization to a safe shutdown condition.

Control room operators completed an orderly shutdown and achieved Mode 5 (RCS temperature less than 200 degrees F) at 1:57 p.m. The UE was terminated at 2:05 p.m. The inspectors determined that operators properly responded to indications of an RCS leak inside containment, appropriately implemented the emergency plan, and promptly stabilized the plant in a safe shutdown condition.

Immediately following event termination, the inspectors reviewed the event’s risk significance with licensee risk analysts and the NRC regional senior risk analyst. Throughout the event, the RCS leakrate remained a small fraction of the normal charging system make-up capacity. This leakrate in turn, provided operators with sufficient time to perform an orderly plant shutdown and cooldown. The inspectors determined that the increase in conditional core damage probability for this event was minimal, and that no additional NRC reactive response was necessary.

On December 15, a 5 gpm RCS leak was identified following plant restart from this event. Operators shut down and cooled down the plant to investigate and correct the cause of the leak. The inspectors reviewed the RCS leak events and monitored the subsequent restart to determine whether station personnel properly evaluated and corrected the cause of the RCS leakage. Inspection activities included a conference call on December 17, between NRC Region I, NRC NRR, and licensee management to discuss the cause of the second RCS leak and corrective action plans.

b. Findings

The inspectors identified a Non-Cited Violation for inadequate corrective actions to restore RCS barrier integrity. Station personnel did not fully understand the effects of their corrective maintenance or the design of the valve packing configuration prior to restarting the unit on December 15. The leak from 2RCS-557B was not fully corrected and mechanics inadvertently damaged packing on three RCS vent and drain valves. Consequently, 2RCS-557B valve packing failed during adjustment on December 15, and a second plant shutdown was performed to effect repairs.

The source of the December 11, RCS leak was system leakage past the packing gland of a ‘B’ RCS loop two inch drain valve (2RCS-MOV557B). An event response team (ERT) determined that the apparent cause of the leak was intergranular stress corrosion cracking (IGSCC), which caused one of two packing gland eyebolts to break (CR 00-4296). The eyebolt material was susceptible to IGSCC due to its hardness value and environment (temperature and humidity). Mechanics repacked 2RCS-MOV557B and replaced the packing gland eyebolts on six similar valves. Three of these valves (including 2RCS-MOV557B) exhibited continued packing gland leakage during the post maintenance test at full RCS pressure. Station personnel were confident that readjusting the valve packing, using increased torque, would stop the leakage. On December 15, a reactor startup was performed while awaiting readiness to perform the packing adjustments. When mechanics attempted to increase the packing gland torque on 2RCS-557B, the valve packing blew out. This action initiated a 5 gpm identified RCS leak and necessitated a reactor shutdown.

A second ERT determined that inadequate valve packing consolidation on 2RCS-557B following the December 12, packing replacement and excessive stress applied to packing during eyebolt replacements on December 14, caused the RCS leakage observed on December 15 (CR 00-4372). The ERT identified numerous valve packing program deficiencies. The ERT report identified reasonable corrective actions to address the causes of the second 2RCS-557B packing leak. The inspectors determined this event revealed knowledge, work instruction, and work practice deficiencies associated with the station’s valve packing program implementation. Specific performance deficiencies included the following:

- The packing configuration for 2RCS-557B (and 3 additional RCS vent and drain valves) had been modified from conventional braided rope style to composite style during the last refueling outage. Station personnel were unaware that the composite packing configuration was fragile and vulnerable to failure if installed incorrectly. The likelihood of packing damage during packing gland eye bolt replacement was not considered.

- Mechanics unknowingly damaged (crushed) valve packing on four valves (including 2RCS-557B) during packing gland eye bolt replacement. Their method of using wedges and hammers caused excessive point loading beyond the design of the composite packing.

- Following replacement, 2RCS-557B packing was not properly consolidated prior to startup.

- Mechanics did not install a top carbon bushing typical for the composite type packing for three valves (including 2RCS-557B) or identify the configuration problem.

The inspectors discussed the second 2RCS-557B packing leak with station management and noted that the organization had missed opportunities to identify the problems listed above prior to unit restart. The senior management team acknowledged this observation and initiated CR 01-0059 to identify additional lessons learned from this event.

This ineffective corrective action issue was more than minor because it had an actual impact on safety in that the reactor coolant system barrier was degraded during power operation. The issue was evaluated using the phase 1 SDP for the barrier integrity cornerstone. The finding did not create an open pathway in the physical integrity of the reactor containment or adversely affect the ability to control containment pressure. Therefore, a phase II SDP evaluation was not warranted. Based on this phase 1 SDP analysis, the inspectors determined the event had very low safety significance and was a GREEN finding.

10 CFR 50, Appendix B, Criterion XVI “Corrective Actions,” requires in part that measures be established to assure conditions adverse to quality are promptly identified and corrected. Contrary to above, corrective actions to resolve excessive packing leakage from 2RCS-557B were ineffective. Specifically, on December 15, 2000, following corrective maintenance to repack 2RCS-557B, the valve packing blew out and initiated a 5 gpm leak past the RCS barrier. The violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A of the Enforcement Policy, issued May 1, 2000 (65 FR 25368). **(NCV 05000412/00-12-01)**

.2 <u>(Closed) Licensee Event Report (LER) 05000334/2000-006</u>: Reactor Trip/Turbine Trip Due to Turbine EH Loss of Control Power. This event was discussed in NRC Inspection Report Nos. 50-334(412)/00-06. No new issues were revealed by the LER. This LER was closed during an onsite review.

4OA5 <u>Other</u>

.1 <u>(Closed) Inspector Follow-up Item (IFI) 50-334/1999-010-01</u>: Potential Licensed Operator Requalification Exam Security Compromise.

Based on the adequacy of the licensee’s evaluation, corrective actions, and the information contained in CR 99-3236, this issue is closed.

4OA6 Management Meetings

.1 Regional Administrator Visit

On December 14 and 15, Mr. Hubert Miller, NRC Region I Administrator, Mr. Wayne Lanning, Director, Region I Division of Reactor Safety, and Mr. John Rogge, Chief, Reactor Projects Branch 7 visited the site. They met with various FirstEnergy senior management including Mr. Robert F. Saunders, President and Chief Nuclear Officer of FirstEnergy and Mr. Lew Myers, Senior Vice President, Beaver Valley Nuclear Power Station. Mr. Miller spoke to an assembly of Beaver Valley Power Station employees and toured the facility with other NRC staff members.

.2 Exit Meeting Summary

The inspectors presented the inspection results to Mr. Lew Myers and other members of licensee management following the conclusion of the inspection on January 5, 2001. The licensee acknowledged the findings presented.

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

4OA7 Licensee-Identified Violations

The following finding of very low 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 NCV.

NCV Tracking Number

(1) NCV 05000412/2000-012-02

Requirement Licensee Failed to Meet

Technical Specification 6.11 requires that procedures shall be prepared consistently with 10 CFR Part 20 and shall be adhered to for all operations involving personnel radiation exposure. Radiological Procedure 8.3, “Radiation Barrier Key Control,” Rev. 3 requires that doors, posted as Locked High Radiation Areas (LHRAs), be closed and locked. Contrary to the above, on December 13, 2000, a radiation technician, while performing the required high radiation area barrier checks, found a Unit 2 LHRA door at the base of the stairwell behind containment elevator (R-92-2) closed, but not locked. The area was surveyed, and no accessible areas had dose rates which exceeded 1,000 millirem per hour at 30 centimeters. The highest dose rate in the area was 900 millirem per hour on contact. This event was entered into the licensee’s corrective action system as CR 00-4327.

PERSONS CONTACTED

Licensee:

L. Myers	Senior Vice President, FENOC
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. Sipp	Manager, Health Physics
R. Freund	Supervisor, Unit 2 Radiological Operations
D. Girdwood	Supervisor, Unit 1 Radiological Operations
J. Lebda	Supervisor, Radiological Engineering and Health
C. Brooks	Plant Services Director, Acting
G. Davie	Training Manager
P. Schwartz	Operations Training Superintendent
T. Kuhar	Training Supervisor

ITEMS OPENED, CLOSED AND DISCUSSED

Opened/Closed

05000412/2000-012-01	NCV	Failure to Adequately Perform Corrective Maintenance to Resolve Packing Leakage from 2RCS-557B. Violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action.” (Section 4OA3.1)
05000412/2000-012-02	NCV	Failure to Control Locked High Radiation Area Door. Violation of Technical Specification 6.11. (Section 4OA7)

Closed

05000334/1999-010-01	IFI	Potential Licensed Operator Requalification Exam Security Issue. (Section 4OA5)
05000334/2000-006	LER	Reactor Trip/Turbine Trip Due to Turbine EH Loss of Control Power. (Section 4OA3.2)

LIST OF ACRONYMS USED

ADAMS	NRC's Document System
AFW	Auxiliary Feedwater
BCO	Basis for Continued Operation
CFR	Code of Federal Regulations
CR	Condition Report
DEP	Drill/Exercise Performance
dpm	drops per minute
EOP	Emergency Operating Procedure
EPIP	Emergency Plan Implementing Procedure
EPP	Emergency Preparedness Plan
ERT	Event Response Team
FENOC	FirstEnergy Nuclear Operating Company
gpm	Gallons Per Minute
IFI	Inspector Follow-up Item
IGSCC	Intergranular Stress Corrosion Cracking
LER	Licensee Event Report
LHRA	Locked High Radiation Area
MR	Maintenance Rule
MSP	Maintenance Surveillance Procedure
MSSV	Main Steam Safety Valve
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NOP	Normal Operating Procedure
NRC	Nuclear Regulatory Commission
NSS	Nuclear Shift Supervisor
NSSS	Nuclear Safety System Supplier
NUREG	NRC Technical Report Designation (Nuclear Regulatory Commission)
OPΔT	Over Pressure Delta Temperature
OST	Operating Surveillance Test
OTΔT	Over Temperature Delta Temperature
PARS	Publicly Available Records
PCM	Personnel Contamination Monitor
PM	Portal Monitor
PMP	Preventive Maintenance Procedure
PMT	Post-maintenance Test
RCS	Reactor Coolant System
RIP	Radiological Instrument Procedure
RP	Radiological Procedure
SCBA	Self-Contained Breathing Apparatus
SDP	Significance Determination Process
SG	Specific Gravity
SSC	Systems, Structures, and Components
TER	Technical Evaluation Report
TLD	ThermoLuminescent Dosimeter
TS	Technical Specification
TSC	Technical Support Center
UE	Unusual Event

ATTACHMENT 1

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.