July 27, 2000

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

SUBJECT: NRC’S BEAVER VALLEY INTEGRATED INSPECTION REPORT 05000334/2000-005; 05000412/20000-005

Dear Mr. Myers:

On July 1, 2000, the NRC completed an inspection at the Beaver Valley 1 & 2 reactor facilities. The enclosed report presents the results of that inspection. The results of this inspection were discussed on July 7, 2000, with you and other 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. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC identified two findings that were elevated under the risk significance determination process and were determined to be of very low safety significance (Green). One of the findings involved an inoperable over temperature delta temperature channel and was a violation of NRC requirements. This violation is being treated as a non-cited violation (NCV), consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368)." These issues have been entered into your corrective action program and are discussed in the summary of findings and in the body of the enclosed inspection report. If you contest the violation or severity level of the NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-001, with copies to the Regional Administrator, Region I, and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001.

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We appreciate your cooperation. Please contact me at 610-337-5146 if you have any questions regarding this letter.

Sincerely,

/RA/

John F. Rogge, Chief
Projects Branch 7
Division of Reactor Projects

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

Enclosure:
1. Inspection Report 05000334/2000-005; 05000412/2000-005

cc w/encl:
K. Ostrowski, 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
W. Pearce, Manager, Operations
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

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REGION I

Docket Nos. License Nos.	05000334, 05000412 DPR-66, NPF-73
Report Nos.	05000334/2000-005, 05000412/2000-005
Licensee:	FirstEnergy Nuclear Operating Company
Facility:	Beaver Valley Power Station, Units 1 and 2
Location:	Post Office Box 4 Shippingport, PA 15077
Dates:	May 14, 2000 through July 1, 2000
Inspectors:	D. Kern, Senior Resident Inspector G. Dentel, Resident Inspector G. Wertz, Resident Inspector J. McFadden, Health Physicist
Approved by:	J. Rogge, Chief Projects Branch 7 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000334-00-05, IR 05000412-00-05, on 05/14-07/01/2000; FirstEnergy Nuclear Operating Company; Beaver Valley Power Station; Units 1 & 2. Maintenance Rule Implementation and Event Follow-up.

The inspection was conducted by resident inspectors and a regional health physics inspector. The inspection identified two green issues, one of which was a non-cited violation, and a cross-cutting issue which was assigned no color. The safety significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process (SDP) in Inspection Manual Chapter 0609 (see Attachment 1).

Cornerstone: Mitigating Systems

- **Green** Inadequate maintenance resulted in additional out of service time for the risk significant Unit 1 “A” auxiliary river water pump. Additional performance deficiencies identified included untimely post maintenance testing and insufficient operator awareness of risk and configuration management.

  The finding was determined to have very low safety significance, because redundant mitigating equipment was available during the period that this pump was out of service for maintenance. No violations of NRC requirements were identified. (Section 1R12.2)

- **Green** A human error during an instrument calibration resulted in an inoperable Over Temperature Delta Temperature reactor protection system channel, which exceeded the technical specification allowed out of service time.

  The finding was determined to have very low safety significance as the remaining channels were operable and were available to provide the necessary protective trip signals. Failure to place the inoperable channel’s bistable in trip within 6 hours was a non-cited violation of technical specification 3.3.1.1, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000. (Section 4OA3.1)

Cross-cutting Issues: Problem Identification and Resolution

- **No Color** On two occasions, problem assessments did not properly evaluate potential risk significance and implement timely effective corrective actions. Although these deficiencies were not the root or contributing causes to the actual events, they represent adverse performance which limited the licensee’s ability to identify and correct adverse safety conditions. Specifically, 1) station personnel did not recognize the potential risk significance of the degraded “A” auxiliary river water pump seal and did not correct the condition in a timely manner; and 2) the safety significance assessment for a reactor protection system (RPS) miscalibration event was also deficient, in that engineers incorrectly concluded that protective functions of the instrument channel were not affected.

Additionally, corrective actions for the RPS miscalibration event did not preclude two repeat miscalibration occurrences. (Section 4OA4)

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Report Details

SUMMARY OF PLANT STATUS: Unit 1 began this inspection period at 100 percent power. The unit remained at or near full power except for a power reduction on May 27 to 60 percent power to replace a leaking seal on the “B” main feedwater pump. The unit returned to full power operation on May 30. Unit 2 began this inspection period at 100 percent power and remained at or near full power throughout the inspection period.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

Hot weather and corresponding warm river water (RW) temperatures pose challenges to various mitigating systems and increase the likelihood of initiating events. In 1999, hot weather challenged the Unit 1 station chillers causing degraded containment temperature control which approached technical specification (TS) limits. During the same period, Unit 2 was forced to shut down due to marine fouling of the service water (SW) system which led to a degraded emergency diesel generator.

The inspectors reviewed Unit 1 station chiller performance and discussed planned chiller upgrades with the system engineer to determine whether appropriate actions were implemented to maintain containment temperature control as required by TS. Additionally, the inspectors reviewed recent SW system biocide treatment results, an Independent Safety Evaluation Group assessment of the biofouling control program, and the most recently performed 1/2OST-30.19, “Main and Alternate Intake Structure Silt Check and Bay Cleaning,” Rev. 8, to determine whether appropriate measures were in place to monitor and maintain SW system performance during hot weather conditions.

Station personnel proposed a Unit 2 cooling tower chemical cleaning and a Unit 1 chiller upgrade during this inspection period. Each activity had the potential to cause a plant transient if not properly implemented. The inspectors reviewed the plans and controls established for the cooling tower chemical cleaning evolution. The proposed cleaning was subsequently canceled. The Unit 1 chiller upgrades were scheduled for the July-September 2000 period. The inspectors reviewed the implementation plan to determine whether the transition plan from the existing chillers to the new chillers addressed the potential for loss of chillers during this hot weather period.

b. Issues and Findings

There were no findings identified.

1R04 Equipment Alignments

a. Inspection Scope

The inspectors performed a complete system walkdown of the Unit 1 high head safety injection (HHSI) system and partial system walkdowns of the Unit 1 RW and Unit 2 standby service water (SWE) systems. The inspectors reviewed the system alignment as described on plant drawings and performed field verification of major equipment alignment. The inspectors also reviewed the outstanding work orders, condition reports, and engineering questions/evaluations on the Unit 1 HHSI system.

b. Issues and Findings

There were no findings 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 and 2 normal switchgear rooms;
- Unit 1 cable spreading area;
- Unit 1 turbine building and oil storage room; and
- Unit 2 west cable vault area.

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. Issues and Findings

There were no findings identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed Unit 1 and 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. Issues and Findings

There were no findings identified.

1R12 Maintenance Rule Implementation

.1 Unit 2, 125 Volt 2-2 Station Battery

a. Inspection Scope

On June 6, 2000, during a periodic TS surveillance test, 2-2 station battery cell 33 voltage measured 2.09 volts which placed the battery in an operable, but degraded condition. Technical specifications required corrective action to restore cell 33 voltage to $\geq$ 2.13 volts within the following 6 days. Engineers had already been monitoring the 125 volt station battery system as a maintenance rule category (a)(1), high safety significance system, due to previous failures of the 2-1 station battery which had forced a plant shutdown in 1998. The inspectors reviewed 2-2 station battery performance history, maintenance rule system categorization, train specific performance goals, causal assessment, corrective maintenance effectiveness, and contingency planning for potential continued battery degradation. An equalizing charge successfully restored cell 33 voltage to above 2.13 volts on June 9. Following the equalizing charge, three additional cells (41, 45, and 47) indicated reduced voltages, but remained above the 2.13 volt TS action limit. Station battery performance monitoring frequency was increased, and five temporary modifications were developed as contingency actions to improve the organization's readiness to quickly restore operability in the event that the 2-2 station battery continued to degrade.

b. Issues and Findings

There were no findings identified.

.2 Ineffective Maintenance on the "A" Auxiliary River Water Pump

a. Inspection Scope

During routine review of the operators logs, the inspectors noticed that Unit 1 entered into a high probabilistic risk assessment (PRA) configuration with an instantaneous core damage frequency (CDF) of 1.8E-4 on April 23, due to an apparent failure of the "A" Auxiliary River Water (ARW) pump during a surveillance test. The inspectors reviewed the pump failure mechanism, corrective maintenance, maintenance rule impact, and associated risk significance.

b. Issues and Findings

Background

On April 2, the “A” ARW pump packing began to smoke profusely during the performance of a routine surveillance test. The pump was declared inoperable. The packing was replaced the following day and the surveillance repeated on April 11. During this test, the packing again began to smoke. In addition, local temperature measurements on the packing housing indicated a temperature of 469°F and rising. The pump was shut down. That same day, maintenance technicians loosened the packing in order to establish cooling flow. A 2-hour maintenance run of the pump was performed during which time the pump operated satisfactorily. The surveillance was re-performed on April 22, and the packing again began to smoke. Local temperature measurements indicated a temperature of 480°F and rising on the pump packing housing. The pump was shutdown and extensive maintenance was performed on the packing assembly. The “A” ARW pump was successfully post-maintenance tested on April 29.

Risk Significance Assessment and Maintenance Rule Unavailability

The system engineer originally considered the pump unavailable from February 8 until the post-maintenance test was successfully completed on April 29. The inspectors screened the finding using the significance determination process (SDP) phase I which directed further evaluation using phase II. The SDP phase II does not model the ARW system. Therefore, the inspectors reviewed the event using the Beaver Valley Unit 1 Probabilistic Risk Assessment model and consulted the NRC regional Senior Risk Analyst. Given the long duration that the pump was considered unavailable, the subsequent calculated delta Core Damage Frequency (△CDF) resulted in a potential WHITE finding (△CDF>1E-6).

In response to the inspectors’ questions, a maintenance engineer evaluated the condition of the packing and determined that the pump would have been capable of performing its safety function. The inspectors reviewed the availability assessment and concluded that, although the assessment was not rigorous or detailed, the safety function of the pump was maintained.

Based on the new safety function assessment, the system engineer recalculated the maintenance rule unavailability hours and removed the previously assigned maintenance preventable functional failures from the ARW system. The inspectors reevaluated the event and determined that the associated risk significance was low (See Inspectors’ Assessment below). Given the new assessment of the pump’s availability, the maintenance rule application was appropriate.

Inspectors’ Assessment

Maintenance performed on April 3 (packing replacement) and then again on April 11 (packing adjustment) failed to correct the problem with the packing. As a result of this inadequate maintenance, additional out-of-service time of approximately 70 hours (from 9:02 p.m. on April 25 until 6:38 p.m. on April 28) resulted due to the pump being placed under clearance. Using the licensee’s PRA model, this resulted in a △CDF of 1.8 E-7.

While this was a credible impact on plant risk, the risk increase was below the 1E-6 threshold and therefore resulted in a GREEN finding. However, the inspectors also noted additional performance deficiencies as described below.

- Post Maintenance Testing Timeliness

Performance of the post-maintenance tests (PMT) following maintenance that was performed on April 3 and again on April 11 was not done in a timely fashion given the risk significance of the ARW system. The PMT's were not performed until 7 and 11 days, respectively, following completion of the maintenance. During this second time period, the "A" main intake bay was removed from service for 3 days for silt inspection and cleaning. This work activity rendered the "A" RW pump unavailable while the bay was drained of RW. Although the operability assessment ultimately determined that the "A" ARW pump remained available despite its degraded seal, the potential existed for additional safety related equipment (e.g. "A" RW pump) to be removed from service prior to demonstrating that redundant risk significant equipment (e.g. "A" ARW pump) was available.

- Risk Awareness and Risk Configuration Management

Nuclear Shift Supervisors (NSS), on two occasions, by allowing the "A" ARW pump PMTs to be performed several days after completion of the work, did not demonstrate appreciation of the risk significance of the pump and its impact to scheduled maintenance. Based on the licensee's PRA, when an ARW pump is out of service, the station's instantaneous CDF increases to 1.08 E-4. The daily risk assessment performed by the PRA engineers assumes that equipment is available when the clearance is removed. Other equipment is then scheduled to be removed from service for maintenance. Station risk planning and risk management does not address delayed PMT's for risk significant equipment. The licensee documented this in condition report (CR) 002261.

.3 Other Maintenance Rule Inspection Items

a. Inspection Scope

The inspectors reviewed the implementation of the Maintenance Rule, including evaluations for maintenance preventable function failures and challenges to plant level and specific system performance criteria. The following issues were evaluated:

- Unit 1 4 kilovolt (kV) Bus "C" feeder breaker 4KVS-1C-1C6 failure to close during the 13th refueling outage(1R13); and
- Unit 1 manual reactor trip resulting from an auxiliary steam valve failing to close and subsequent degradation of condenser vacuum.

b. Issues and Findings

There were no findings identified.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

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:

- Planned preventive maintenance performed on the Unit 2 "B" HHSI pump. This item was selected due to the risk significance of the HHSI pump and that the scheduled out-of-service time was extended from 4 days to 9 days during the week before the maintenance was to be completed. This scheduled out-of-service time extension was due to other planned maintenance and manpower constraints.

- Unplanned repairs performed on the wiring associated with the power supply breaker (MCC-2-E05-4F) for a Unit 2 primary component cooling water header isolation valve. A maintenance technician using a questioning attitude had identified the damaged wiring during an earlier maintenance activity. (CR 00-1959)

- Realignment of the onsite to offsite 4kV system due to the discovery of the cause of the "C" bus feeder breaker 4KVS-1C-1C6 failure to close during the 1R13 outage. This change was to align the onsite 4kV supply system to the dedicated offsite supply. This change eliminated the risk associated with a possible failure of any additional 4kV breakers to close on a loss of normal power (main generator) and subsequent automatic transfer of the 4kV system. (CR 00-1733)
- Planned repair performed on a steam leak in the downstream piping of the Main Steam Reducer, PCV-1AS-100. The failure of the actuator packing on this valve resulted in a manual reactor trip on April 17. This repair involved installation and sealant injection into an external piping clamp.

- Planned replacement performed on the Unit 2 SWE traveling water screen (2SWE-SSC1) filters. This maintenance was planned and implemented such that the traveling water screen remained available to support operation of the SWE if necessary.

- Planned maintenance of the Emergency Response Facility (ERF) diesel generator. The ERF diesel is non-safety related but is risk significant as evaluated in the plant's PRA. It provides a backup power supply to the station and containment instrument air compressors as well as the startup feedwater pump. The inspectors reviewed the risk assessment performed for the ERF diesel maintenance activity.

b. Issues and Findings

There were no findings identified.

1R14 Personnel Performance During Non-routine Plant Evolutions

a. Inspection Scope

The inspectors reviewed operator performance during the following non routine plant evolutions:

- On June 4, during an inspection activity where previous fuel cycle assemblies were being partially disassembled and inspected in the Unit 1 spent fuel pool, a fuel rod separated into two pieces while being removed from the assembly. The inspectors reviewed the NSS' assessment and actions including entry into and exit from Abnormal Operating Procedure 1OM-53C.4.1.49.1, Irradiated Fuel Damage, Rev. 3. (CR 001930)

- On June 20, an inadvertent HALON discharge (fire protection system actuation) occurred in the control building. The inspectors observed the operators' responses to the actuation, reviewed the classification of the event, and examined the root cause. No actual fire had occurred. The cause of the event was attributed to inadequate human performance associated with procedural adherence. (CR 002103)

- On June 16, an unexpected change in pressurizer pressure control pressure occurred automatically. The inspectors reviewed operator response and automatic actions associated with the event (backup pressurizer heaters energizing). There was no safety significance to the event since the pressurizer pressure remained in an acceptable band throughout the event. (CR 002070)

- Human performance errors during a Unit 2 Over Temperature Delta Temperature (OTDT) instrument calibration and a Unit 1 emergency diesel generator surveillance led to reportable events discussed in sections 4OA3.1 and 4OA3.2.

b. Issues and Findings

There were no findings 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:

- Unit 1 Steam Line Break Increased Projected Primary-to-Secondary Leakage Basis for Continued Operation (BCO) 1-00-004;

- Unit 1 “A” RW pump low pump head (CR 001725);
- Unit 1 “B” charging pump high seal flow (CR 002081);
- Twenty-one overdue Unit 1 and Unit 2 preventive maintenance tasks. This review was performed due to previous problems identified in the preventive maintenance program (see NRC Integrated Inspection Report Nos. 05000334(412)/1999010); and,
- Unit 2 “B” and “C” SW pumps’ low pump head ratio (CRs 002182 and 001843).

b. Issues and Findings

There were no findings 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.

- 1OST-47.3, “Containment Isolation and ASME Section XI Test,” Rev. 25, for the “C” Main Feedwater Bypass Flow Control Valve [FCV-1FW-499] following maintenance (actuator replacement).
- 1OST-30.1B, “[1WR-P-9B] Auxiliary River Water Pump Test,” Rev. 19, following preventive maintenance (motor lubrication, vacuum break check valve inspection).
- 1OST-30.3, “Reactor Plant River Water 1B Test,” Rev. 22, following maintenance to repair a motor oil cooling line leak.
- 1OST-30.1A, “[1WR-P-9A] Auxiliary River Water Pump Test,” Rev. 20, following maintenance on the packing assembly. For more information on this pump, see Section 1R12.2.
- 2OST-24.3, “Motor Driven Auxiliary Feed Pump [2FWE*P23B] Test,” Rev. 22, following planned preventive maintenance (oil change and pump repack).
- 2OST-36.1, “Emergency Diesel Generator [2EGS*EG2-1] Monthly Test,” Rev. 28, following maintenance on 2SWS-408 (leak).
- 1/2OST-58E.1, ”RG Diesel Generator [RG-EG-1] Test,” Rev. 18, following maintenance which replaced the ERF diesel generator voltage regulator potentiometer and repaired the cooling fan motor and various oil leaks.
- 1/2CM-75-BAT-1E, “Station Battery Replacement,” Rev. 2 and work order (WO) WO-00-003566, “UPS #1 Battery for ERF Substation” following replacement of the #3 battery for the ERF diesel generator.

b. Issues and Findings

There were no findings 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.

- 1OST-30.1B — "[1WR-P-9B] Auxiliary River Water Pump Test," Rev. 19
- 1OST-36.1 — "Diesel Generator No. 1 Monthly Test," Rev. 26
- 2OST-1.1 — "Control Rod Assembly Partial Movement Test," Rev. 3
- 2MSP-37.04-E — "2P 480V Emergency Bus Degraded Voltage Relays 27-RP200AB/BC," Rev. 11

  2MSP-36-37-E — "2DF 4kV Emergency Bus Degraded Voltage Relays 27-VF3200AB/BC," Rev. 10

b. Issues and Findings

There were no findings identified.

2. RADIATION SAFETY

Cornerstones: Occupational Radiation Safety, Public 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 examined the adequacy and compliance with regulatory requirements and TSs for posting, barricading, and locking entrances to high radiation areas (HRAs) and locked high radiation areas (LHRAs) in Units 1 and 2. A walkdown of each of these locations, including the Primary Auxiliary Building (PAB) and the Condensate Polishing Building (CPB), was performed.

Unit 1	Letdown cubicle, Primary Auxiliary Building PAB-722	HRA
	East valve trench, PAB-722, north gate	LHRA
	Blender cubicle, PAB-722	HRA
	East valve trench, PAB-722, south gate	LHRA
	Boron recovery evaporator hold tank cubicles, PAB-722	HRA
	Low level waste tank cubicle, PAB-735	HRA
	High level waste tank cubicle, PAB-735	HRA
	Entrance to degasifier cubicle B, PAB-735	HRA
	Boric acid tank cubicle 1A, PAB-752	HRA
	Volume control tank cubicle, PAB-752	LHRA

	Catwalk above LW-1-2, decontamination bldg. 735	LHRA
	Solid waste, east wall	LHRA
	Solid waste, north wall	LHRA
Unit 2	Volume control tank, PAB-755	LHRA
	Degasifier cubicle, PAB-718	HRA
	Letdown heat exchanger, PAB-718	HRA
	Liquid waste drain tanks, PAB-710'6"	HRA
	Ladder to above drum storage, CPB-735	LHRA
	Ladder to high integrity container storage, CPB-735	LHRA
	Drum storage room, CPB-735	LHRA

In this area, the inspection also included a review of the Health Physics (HP) Manual procedures, Radiation Protection (RP) 8.3 Radiation Barrier Key Control and RP 7.13 Barrier Checks.

The inspectors witnessed the pre-job briefing for a subatmospheric containment entry on Radiation Work Permit (RWP) 100-1043 to replace a switch on an air compressor on the morning of June 28, 2000.

The inspection included an evaluation of the effectiveness and timeliness of the licensee's response to the following high radiation area radiological incidents which occurred in cubicles in the reactor building containment, a steam generator primary manway opening, and the reactor cavity transfer canal: CRs. 00-0773, 00-0849, 00-1106, and 00-1273. The review included an evaluation of the associated cause evaluations and corrective actions.

The inspectors reviewed sixteen CRs that addressed worker (00-0431, 00-0652, 00-0728, 00-0935, 00-0971, 00-0123, 00-1165, 00-1234, 00-1303, 00-1323, and 00-1344) and/or radiation protection technician (00-0701, 00-1017, 00-1082, 00-2020, and 00-2072) performance errors, occurring between January and June 2000. The above-cited CRs addressed lack of proper compliance with RWPs, with RWP briefings, with radiological posting requirements, and with radioactive waste reduction practices. The review included an evaluation of the associated cause evaluations and corrective actions.

b. <u>Issues and Findings</u>

There were no findings 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) radiation exposure planning and control.

The inspectors reviewed the radiological work planning for the following five high exposure jobs completed during the last refueling outage (Unit 1, first quarter 2000) by examining the draft ALARA outage report and the RWP and ALARA Review (AR) records for these jobs. In this area, pre-job, on-going-job, and post-job ALARA reviews, RWP requirements, and radiation surveys were examined.

- RWP 100-4032/AR 00-1-08 Reactor disassembly and reassembly
- RWP 100-4040/AR 00-1-14 Steam generator foreign object search and retrieval
- RWP 100-4043/AR 00-1-16 Steam generator channel head work
- RWP 100-4066/AR 00-1-34 Design Change Package 2282 Install permanent cavity seal
- RWP 100-4069/AR 00-1-36 Steam generator tube sleeving

In this area, the inspection also included a review of the outage ALARA reports for the Unit 2 seventh refueling outage and for the Unit 1 1999 surveillance outage.

The inspectors reviewed the assumptions and basis for the current annual exposure estimate and annual exposure goal by performing the following activities.

Review of the following procedures:

- Nuclear Power Division Administrative Manual Procedure 3.1 Exposure Control
- HP Manual Appendix 11 ALARA Program
- HP Manual Procedure RP 8.5 ALARA Review Program
- HP Manual Procedure RP 8.11 Respiratory Protection ALARA Evaluation

Discussions with the Radiation Protection Manager and the ALARA Health Physicist; and
Attendance at the Nuclear ALARA Review Committee Meeting on June 27, 2000.

The inspectors investigated the exposure tracking system, its level of exposure tracking detail, exposure report timeliness, and exposure report distribution by performing the following activities.

Discussions with the Radiation Protection Manager and the ALARA Health Physicist; and

Review of the exposure report dated June 23, 2000 (0728 hours), which contained RWP total person-rem, person-hours, and average dose rates (actual and budgeted) and occupational radiation exposure reports by department and by craft.

The inspection also included a review of the Health Physics Program Audit No. BV-C-99-14 (August 19 - October 4, 1999) by the Quality Services Unit which included the ALARA program in its scope. This audit resulted in three CRs and two recommendations.

b. Issues and Findings

There were no findings identified.

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

The inspection included the following activities to determine the effectiveness of radiation monitoring instrumentation.

The inspectors reviewed the radiation survey meter qualification training and the process for authorization to be issued a radiation survey meter for use in the plant.

b. Issues and Findings

There were no findings identified.

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

The inspection included the following activities to determine the effectiveness of radioactive material processing and transportation:

Radioactive waste system walkdown

The inspection included a walkdown of accessible portions of the station's radioactive liquid and radioactive solid waste collection, processing, and storage systems/locations to verify that the current system configuration and operation agreed with descriptions contained in the Updated Final Safety Analysis Report and in the Process Control Program (PCP).

The areas reviewed during the walkdown included: the auxiliary building, fuel building, decontamination building, solid waste building, and turbine building in Unit 1; the auxiliary building, fuel and decontamination building, solid waste building, condensate polishing building, and turbine building in Unit 2; the common trailer storage within the protected area; and the waste handling building, and the outside warehouse.

The inspectors accomplished a selective review of the following items during the walk-down:

- The status of non-operational or abandoned in-place radioactive waste process equipment and administrative and physical controls for the systems.
- Any changes made to radioactive waste processing systems and potential radiological impact, specifically the updated version of the software program used for waste scaling factors.
- The current processes for transferring radioactive waste resin and sludge to shipping containers and mixing and sampling of the waste, specifically operational procedures for resin sluicing and for resin dewatering.

Waste characterization and classification

In this area, the inspection included a selective review of the following items:

- The radio-chemical sample analysis results for radioactive waste streams including the PCP (Issue 5.0, Rev. 1, effective date 09-22-98) and the Radiological Engineering Administrative Manual, Part 4 Radwaste, Procedure 4.107, “10 Code of Federal Regulations (CFR) 61 Sample Analysis Program.”
- The development of scaling factors for difficult to detect and measure radionuclides using Procedure 4.107.
- The methods and practices to detect changes in waste streams as described in the PCP.
- The methods and practices to determine waste classification (10 CFR 61.55) and to determine Department of Transportation (DOT) shipment subtype (49 CFR 473), specifically Health Physics Manual Chapter 3, Part 3, “Handling of radioactive or contaminated materials.”

Shipment preparation

In this area, the inspection included the following:

- The observation of remote movement of radioactive waste drums within the drum storage area in Unit 2 for the purpose of weighing and surveying drums.
- A review of the technical instructions presented to workers during routine training which was conducted on a biennial basis.
- Verification that training was provided to personnel involved in resin dewatering and radioactive waste/material shipment activities.

Shipping records

In this area, the inspection involved a review of the following six non-excepted package shipment records for compliance with NRC and DOT requirements:

- Shipment B-2560, waste, Type B cask, High Integrity Container (HIC), Class B, Yellow-III
- Shipment B-2569, radioactive material, wooden box, Surface Containment Object (SCO)-I

- Shipment B-2575, waste, metal box, Class A(U), Low Specific Activity (L.A.)-II
- Shipment B-2589, radioactive material, Type A container, Yellow-II
- Shipment B-2611, radioactive material, DOT 6M drum, Yellow-II
- Shipment B-2629, radioactive material, metal box, SCO-II

Identification and resolution of problems

The inspection included a review of the following audits and self-assessments related to the radioactive material and transportation programs since the previous inspection and a determination if identified problems were entered into the corrective action program for resolution:

- Quality Assurance Audit - Radioactive Waste Management and Transportation Program, April 23 to June 10, 1999, dated June 30, 1999.
- Radioactive Waste Shipping/Transportation Self-Assessment, September 17, 1999.

In this area, the inspection also involved the review of ten CRs written against the radioactive material and shipping program from April 1999 to March 2000 for adequacy and timeliness of corrective actions.

b. Issues and Findings

There were no findings identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification

.1 Unplanned Scrams and Scrams with Loss of Normal Heat Sink

a. Inspection Scope

The inspectors reviewed the performance indicators for unplanned scrams per 7000 critical hours and scrams with loss of normal heat sink for Unit 1 and Unit 2. The inspectors verified accuracy of the reported data through reviews of Licensee Event Reports (LERs) and additional records. The inspectors reviewed one year of data for unplanned scrams and three years of data for scrams with loss of normal heat sink. No problems with performance indicator accuracy or completeness were identified.

b. Issues and Findings

There were no findings identified.

.2 Occupation Exposure Control Effectiveness

a. Inspection Scope

The inspectors selectively examined records used by the licensee to identify occurrences involving high radiation areas, very high radiation areas, and unplanned personnel exposures for the time period from January 1, 2000 to June 15, 2000 against the applicable criteria specified in Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 0, to verify that all conditions that met the NEI criteria were recognized and identified as performance indicators. The reviewed records included corrective action program records and radiologically controlled area access control alarm reports. This examination did not find any problems with the performance indicator accuracy or completeness.

b. Issues and Findings

No findings were identified.

4OA3 Event Follow-up

.1 (Closed) LER 05000334/1999-012-00 and 01: Inoperability of Loop 1 Over Temperature Delta Temperature Function and Resulting Non-Compliance with Technical Specification 3.3.1.1, Table 3.3-1, Action 7, Item a.

a. Inspection Scope

On September 16, 1999, one of the three channels of the Reactor Protection System (RPS) OTDT reactor trip instrumentation was discovered to have been inoperable and the requirements of TS 3.3.1.1, Table 3.3-1, Action 7, Item a. to trip the associated channel bistables were not performed within the required 6 hours. The cause of the inoperable OTDT channel was due to a miscalibration of the N41 neutron power range instrumentation drawer during performance of a routine MSP. The inspectors reviewed the cause of the event and corrective actions.

b. Issues and Findings

The cause of the event was due to human error. The technician performing calibration surveillance 1MSP-2.03-I, “Power Range Neutron Flux Channel N41 Refueling Calibration,” Rev. 8, misread the voltage value while adjusting the N41 detector instrumentation drawer. The miscalibrated OTDT channel was in service for approximately 31 hours before being removed to investigate a problem with the N41 delta flux meter (which occurred as a result of the mis-calibration). It was during this troubleshooting that the detector instrument drawer was discovered to have been incorrectly calibrated.

The event was evaluated using the SDP phase I and determined to constitute a degradation of “Reactivity Control.” Since the channel was inoperable, it represented an actual loss of a safety function of a train for greater than its technical specification allowed out of service time. As such, an SDP phase II evaluation was performed. The only initiating event affected by a degradation in OTDT would be the Anticipated Transient Without Scram (ATWS). To conduct the Phase II evaluation, the affect on initiating event frequency of an ATWS was revised to reflect the degraded condition and then the SDP phase II worksheets were applied as follows. Assuming one reactor trip per year, the industry failure history for one channel of OTDT and a duration of less than 3 days, the estimated likelihood rating of E from IMC 0609, table I resulted. Since all other mitigating equipment in the ATWS worksheet was operable, the minimum mitigation capability was determined to be a “2". Using Table 2 in IMC 0609, an event with an “E” initiating event likelihood rating and a “2" remaining mitigation capability rating would result in a GREEN finding.

Although the miscalibration is considered to have very low safety significance (Green) using the SDP, the failure to recognize that the OTDT Loop 1 channel was miscalibrated, resulted in an inoperable reactor protection channel in excess of TS requirement 3.3.1.1, Table 3.3-1, Action 7, Item a. and, therefore, was a violation of TSs. This violation is considered a non-cited violation, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368). **(NCV 05000334/2000-005-01)**

<u>Inspectors’ Assessment</u>

The inspectors determined that the original LER did not fully address the safety implications of the event. The plant services Director reviewed this concern, agreed that the LER safety implication assessment was deficient, and a supplemental LER was issued to revise the assessment. The inspectors also identified two similar events which occurred on Unit 2 in February and June 2000 (CR’s 00-0939 and 00-2216). As before, nuclear instrumentation calibration was incorrectly performed due to technicians copying reference data incorrectly onto the procedure. The incorrect instrument calibration data was identified by the NSS’s during their review and did not result in inoperable RPS channels nor TS violations. These repeat calibration problems indicated that the original corrective actions were not fully effective. The corrective actions were enhanced in the LER supplement to include additional peer checking techniques as well as self checking and the inspectors determined these additional actions to be appropriate.

.2 <u>(Closed) LER 05000334/2000-004-00:</u> “Inadvertent Engineered Safety Features Actuation Due to Loss of Power to 4kV Emergency Bus.” The cause of this event was attributed to human error made by an electrician during performance of OST 36.3, “Diesel Generator No. 1 Automatic Test.” This LER was a minor issue and was closed during an onsite review.

.3 <u>(Closed) LER 05000334/1999-013-00:</u> “Entry into Technical Specification 3.0.3 Due to Inoperable Rod Position Indicator for Rods G-3 and J-3.” This LER was a minor issue and was closed during an onsite review.

.4 <u>(Closed) LER 05000334(412)/1999-011 and 1999-011-01</u>: Inoperability of Service Water System Train B Due to Deformed Discharge Expansion Joint on In-Service Pump 2SWS*P21C. This event was discussed in NRC Inspection Report Nos. 50-334(412)/99-10 and 00-01, and NRC Notice of Violation letter dated May 3, 2000. No new issues were revealed by the LER. This LER was closed during an onsite review.

.5 <u>(Closed) LER 05000334/2000-03</u>: Failure to Comply with Technical Specifications Due to Inoperability of One Subsystem of the Containment Recirculation Spray (RS)

System. This event was discussed in NRC Inspection Report Nos. 50-334(412)/00-02. This was a violation of TS 3.6.2.2 which requires restoration of the inoperable RS subsystem within 7 days or shutdown to hot standby within the following 6 hours. The inspectors determined that the cause of this event was isolated and the associated risk was minimal. No new issues were revealed by the LER. This failure constitutes a violation of minor significance and is not subject to formal enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This LER was closed during an onsite review.

.6 (Closed) IFI 05000412/1999-007: Battery Capacity for Shutdown for Two Inoperable Chargers. This IFI was administratively closed during an in-office review.

4OA4 Crosscutting Issues

Problem Identification and Resolution Problems

a. Inspection Scope

The inspectors reviewed problem identification, safety significance assessment, and corrective action implementation associated with the degraded "A" ARW pump seal (Section 1R12.2) and an instrument miscalibration event (Section 4OA3.1).

b. Issues and Findings

The inspectors noted ineffective problem assessment during the "A" ARW pump packing event and a deficient safety significance assessment of an OTDT RPS miscalibration event. Although these deficiencies were not the root or contributing causes to the actual events, they represent adverse performance which limited the licensee's ability to identify and correct adverse safety conditions.

During the "A" ARW river water pump packing event, five CRs associated with the "A" ARW pump were generated and reviewed by the management team (CRs 00-1416, 00-1451, 00-1482, 00-1584, and 00-1633). However, the five CRs were assigned to different owners for resolution and had 90 days permitted for investigation. Management review of CRs did not recognize the potential risk significance of the problem nor was timely action taken to collect information and perform an investigation. Following a discussion of the potential risk significance of the issue with the site senior vice president, the separate CR's were grouped together and a formal review of the problem and root-cause assessment was performed.

Licensee event report 05000334/1999-012 did not fully address the safety implications of a RPS miscalibration event. Although the loop 1 OTDT channel was inoperable for 31 hours, the LER stated that the protective functions of the channel were not affected. In addition, corrective actions did not preclude two (2) repeat miscalibration events. A supplemental LER was issued to correct these deficiencies.

4OA6 Management Meetings

.1 Commissioner Merrifield Visit

On May 15, 2000, The Honorable Mr. Jeffrey S. Merrifield, NRC Commissioner, visited the site. He met with various FirstEnergy senior management including Mr. H. Peter Burg, Chairman and Chief Executive Officer of FirstEnergy Corporation, Mr. Robert F. Saunders, President and Chief Nuclear Officer of FirstEnergy Nuclear Operating Company, and Mr. Lew W. Myers, Senior Vice President, Beaver Valley Nuclear Power Station. Commissioner Merrifield toured the facility and spoke to an assembly of the plant's employees.

.2 Exit Meeting Summary

The inspectors presented the inspection results to Mr. Lew Myers and other members of licensee management at the conclusion of the inspection on July 7, 2000. The licensee acknowledged the findings presented.

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

ITEMS OPENED, CLOSED AND DISCUSSED

Opened/Closed

05000334/2000-005-01	NCV	Instrument Miscalibration Results in Inoperable Over Temperature Delta Temperature Instrument Channel and Violation of TS 3.3.1.1. (4OA3.1)

Closed

0500034/1999-012-00	LER	Inoperability of Loop 1 Over Temperature Delta Temperature Function and Resulting Non-Compliance with Technical Specification 3.3.1.1, Table 3.3-1, Action 7, Item a. (A4OA3.1)
0500034/1999-012-01	LER	Inoperability of Loop 1 Over Temperature Delta Temperature Function and Resulting Non-Compliance with Technical Specification 3.3.1.1, Table 3.3-1, Action 7, Item a. (A4OA3.1)
0500034/2000-004	LER	Inadvertent Engineered Safety Feature Actuation Due to Loss of Power to 4kV Emergency Bus. (A4OA3.2)
0500034/1999-013	LER	Entry into Technical Specification 3.0.3 Due to Inoperable Rod Position Indicator for Rods G-3 and J-3. (A4OA3.3)
05000334(412)/1999-011	LER	Inoperability of Service Water System Train B Due to Deformed Discharge Expansion Joint on In-Service Pump 2SWS*P21C (4OA3.4)
05000334(412)/1999-011-01	LER	Inoperability of Service Water System Train B Due to Deformed Discharge Expansion Joint on In-Service Pump 2SWS*P21C (4OA3.4)
05000334/2000-003	LER	Failure to Comply with Technical Specifications Due to Inoperability of One Subsystem of the Containment Recirculation Spray System (4OA3.5)
05000412/1999-007	IFI	Battery Capacity for Shutdown for Two Inoperable Chargers (4OA3.6)

LIST OF ACRONYMS USED

1R13	Unit 1 13th Refueling Outage
ALARA	As Low As Reasonably Achievable
AR	ALARA Review
ARW	Auxiliary River Water
ATWS	Anticipated Transient Without Scram
BCO	Basis for Continued Operation
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CPB	Condensate Polishing Building
CR	Condition Report
DOT	Department of Transportation
ERF	Emergency Response Facility
HHSI	High Head Safety Injection
HIC	High Integrity Container
HP	Health Physics
HRA	High Radiation Area
kV	Kilovolt
L.A.	Low Specific Activity
LER	Licensee Event Report
LHRA	Locked High Radiation Area
MSP	Maintenance Surveillance Procedure
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NSS	Nuclear Shift Supervisor
OST	Operational Surveillance Test
OTDT	Over Temperature Delta Temperature
PAB	Primary Auxiliary Building
PCP	Process Control Program
PMT	Post Maintenance Test
PRA	Probabilistic Risk Assessment
RP	Radiation Protection
RPS	Reactor Protection System
RS	Recirculation Spray
RW	River Water
RWP	Radiation Work Permit
SCO	Surface Contaminated Object
SDP	Significance Determination Process
SW	Service Water
SWE	Standby Service Water
TS	Technical Specification
WO	Work Order
ΔCDF	Delta Core Damage Frequency

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.