June 29, 2001

Mr. L. W. Myers
Senior Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
P. O. Box 4
Shippingport, Pennsylvania 15077

SUBJECT: BEAVER VALLEY UNIT 1 - NRC FIRE PROTECTION INSPECTION REPORT NO. 05000334/2001-005

Dear Mr. Myers:

On June 8, 2001, the NRC completed a triennial fire protection team inspection at your Beaver Valley Power Station, Unit 1. The enclosed report documents the inspection findings which were discussed at an exit meeting on June 8, 2001, with Mr. Saunders, you, and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's regulations and with the conditions of your license. The purpose of the inspection was to evaluate your post-fire safe shutdown capability and fire protection program. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the team identified an issue of very low safety significance (Green) involving a deficiency in the procedure for shutdown of the plant from outside of the control room. This issue involved a violation of NRC requirements; however, because of the very low safety significance and because it was entered into your corrective action program, the NRC is treating it as a non-cited violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this 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, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington DC 205555-0001; and the NRC Resident Inspector at the Beaver Valley Power Station.

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Sincerely,

/RA/

Wayne D. Lanning, Director
Division of Reactor Safety

Docket No. 05000334
License No. DPR-66

Enclosure: NRC Inspection Report 05000334/2001-005

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	05000334
License No:	DPR-66
Report No:	05000334/2001-005
Licensee:	FirstEnergy Nuclear Operating Company
Facility:	Beaver Valley Power Station, Unit 1
Location:	Shippingport, Pennsylvania
Dates:	May 21 - June 8, 2001
Inspectors:	T. Walker, Sr. Reactor Inspector, Division of Reactor Safety D. Kern, Senior Resident Inspector, Beaver Valley L. Scholl, Sr. Reactor Inspector, DRS M. Maley, Reactor Inspector (in training), NRR
Approved By:	James C. Linville, Chief Electrical Branch Division of Reactor Safety

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SUMMARY OF FINDINGS

IR 05000334-01-05, on 05/21- 06/08/01, FirstEnergy, Beaver Valley Unit 1, Fire Protection.

The inspection was conducted by a team composed of regional specialists and a senior resident inspector. The inspection identified one green finding which was a non-cited violation. The significance of most issues is indicated by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply are indicated by "no color" or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- **Green**. The team identified a non-cited violation of 10 CFR 50, Appendix R for failure to have adequate procedures to assure safe shutdown capability. The team found that the procedure for shutdown from outside the control room did not provide adequate direction to promptly verify river water (RW) cooling to the protected emergency diesel generator (EDG). The delay in verifying RW cooling to the running EDG could result in damage to the EDG and a loss of all AC power. The safety significance of this finding was very low because the likelihood of a fire that would cause a loss of all RW and necessitate a shutdown from outside of the control room was small. (Section 1R05.07)

Report Details

Background

This report presents the results of a triennial fire protection team inspection conducted in accordance with NRC Inspection Procedure (IP) 71111.05, “Fire Protection.” The objective of the inspection was to assess whether FirstEnergy Nuclear Operating Company has implemented an adequate fire protection program and that post-fire safe shut down capabilities have been established and are being properly maintained. The following fire areas were selected for detailed review based on risk insights from the Beaver Valley Unit 1 Probabilistic Risk Assessment Individual Plant Examination of External Events (IPEEE), Rev. 0:

- Control Room General Area (CR-1)
- Control Room Process Instrument Room (CR-4)
- Cable Tray Mezzanine (CS-1)
- Normal Switchgear Room (NS-1)
- Primary Auxiliary Building General Area E (PA-1E)

This inspection was a reduced scope inspection in accordance with the September 22, 2000, revision to IP 71111.05, “Fire Protection.” Issues regarding equipment malfunction due to fire-induced failures of associated circuits were not inspected. Criteria for review of fire-induced circuit failures are currently the subject of a voluntary industry initiative. The definition of associated circuits of concern used was that contained in the March 22, 1982, memorandum from Mattson to Eisenhut, which clarified the requests for information made in Generic Letter 81-12.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems

1R05 Fire Protection (71111.05)

.1 Programmatic Controls

a. Inspection Scope

During tours of the facility, the team observed the material condition of fire protection systems and equipment, the storage of permanent and transient combustible materials, and control of ignition sources. The team also reviewed the procedures that controlled hot-work activities and combustibles at the site. This was accomplished to ensure that the licensee was maintaining the fire protection systems, controlling hot-work activities, and controlling combustible materials in accordance with the fire protection program.

b. Findings

No findings of significance were identified.

.2 Passive Fire Barriers

a. Inspection Scope

The team walked down accessible portions of the selected fire areas to observe material condition and the adequacy of design of fire area boundaries, fire doors, and fire dampers. The team reviewed engineering evaluations, as well as surveillance and functional test procedures for selected items. The team also reviewed the regulatory exemptions and NRC safety evaluations associated with fire protection features for Beaver Valley Unit 1. The design and qualification testing for raceway fire barriers were also reviewed and a walk-down of installed barriers was performed for the selected areas. These reviews were performed to ensure that the fire barrier systems met the licensing and design bases.

The team reviewed penetration seal inspection procedures, and selected penetration seal engineering evaluations. The team also compared the penetration seal ratings with the ratings of the barriers in which they were installed. This was accomplished to ensure that the licensee had installed and maintained fire barrier penetration seals in accordance with the design and licensing bases.

b. Findings

Raceway Protection

At Beaver Valley Unit 1, Hemyc wrap is utilized to protect raceways associated with the power supply cable for the CH-P-1B charging pump in the auxiliary building to meet the separation requirements of 10 CFR 50, Appendix R. A raceway enclosure with a one hour fire rating is required to ensure availability of reactor coolant inventory makeup for postulated fires in the auxiliary building. Since Hemyc fire barriers were installed after the effective date of Appendix R, they are required to meet the technical requirements of Appendix R or have appropriate documentation to justify a deviation.

The NRC previously identified issues at the Shearon Harris Nuclear Power Plant (IR 50-400/99-13) regarding the acceptability of the Hemyc fire barrier qualification tests. The NRC Region II office requested the Office of Nuclear Reactor Regulation (NRR) assistance in Task Interface Agreement (TIA) 99-028, dated November 23, 1999, in evaluating the resolution of these issues.

At Beaver Valley, the licensee evaluated the adequacy of the Hemyc fire barrier in design analysis 8700-DEC-0187, “Hemyc Blanket Fire Barrier Configurations, Rev. 0. This analysis, which concluded the installation at Beaver Valley Unit 1 provided a one hour fire resistance capability, was based, in part, on the results of one of the qualification tests discussed in TIA 99-028.

To resolve concerns associated with the use of Hemyc fire barriers, the licensee initiated engineering change package ECP-00025-CS, “Hemyc Fire Wrap Replacement,” to replace the Hemyc installation with a different barrier that has been satisfactorily tested.

The adequacy of the Hemyc barrier at Beaver Valley Unit 1 is unresolved until the NRC makes a determination on its acceptability or until the licensee completes replacement of the Hemyc with an acceptable alternative barrier. **(URI 05000334/2001-005-01)**

.3 Fire Detection Systems

a. Inspection Scope

The team reviewed the adequacy of the fire detection systems in the selected plant fire areas. This included a walk-down of the systems and review of detector location drawings. The team also reviewed regulatory exemptions and NRC safety evaluations associated with the selected fire areas. These reviews were performed to ensure that the fire detection systems for the selected fire areas were installed and maintained in accordance with the design and licensing bases. The team also reviewed fire detection surveillance procedures to determine the adequacy of fire detection component testing.

b. Findings

No findings of significance were identified.

.4 Fire Suppression Systems and Equipment

a. Inspection Scope

The team reviewed the adequacy of the carbon dioxide (CO_2) suppression system in the cable tray mezzanine and the fixed sprinkler systems in the auxiliary building by performing walk-downs of the systems. In addition, the team verified suppression system functionality and the adequacy of surveillance procedure testing by reviewing several completed surveillance procedures. The team also reviewed initial discharge testing, design specifications, and engineering evaluations for the cable tray mezzanine CO_2 suppression system. This review was performed to ensure that the selected fixed suppression systems met their design and licensing bases.

The team walked down selected standpipe systems and portable extinguishers to determine the material condition of manual fire fighting systems. Electric fire pump flow and pressure tests were also reviewed by the team to ensure that the pump was meeting design requirements.

The team inspected the fire brigade’s protective ensembles, self-contained breathing apparatus (SCBA), and various other fire brigade equipment to determine operational readiness for fire fighting.

b. Findings

Deficiencies with the initial testing and maintenance of the cable tray mezzanine CO2 suppression system were identified during the inspection. These deficiencies involved apparent insufficient CO2 soak times during initial discharge testing and an inappropriate administrative limit for CO2 tank volume. Both of these deficiencies were identified by the licensee prior to the inspection; however, at the time of the inspection, the licensee had not fully evaluated the impact of the insufficient soak times on the CO2 suppression system functionality.

The Beaver Valley Unit 1 Updated Fire Protection Appendix ‘R’ Review (UFPARR) describes the cable tray mezzanine CO2 system as an automatic or manual, double shot, total flooding system that conforms to the requirements of NFPA No. 12, “Standard on Carbon Dioxide Extinguishing Systems,” 1973 (NFPA 12-1973). The Updated Final Safety Analysis Report (UFSAR) also indicates that the system has double shot capability. The cable tray mezzanine CO2 system is designed to attain a 52% CO2 concentration in order to suppress deep-seated fires associated with electrical wiring insulation hazards. Design calculations indicate that 8700 lbs of CO2, discharged over approximately 220 seconds, are required to achieve the required CO2 flooding concentration in the cable tray mezzanine.

CO2 Tank Volume

In CR 01-2835, the licensee determined that the minimum tank volume of 30% specified as the surveillance requirement for the 10 ton CO2 storage tank was not sufficient to assure that the cable tray mezzanine CO2 suppression system could perform its design function. Specifically, the licensee determined that a tank volume of approximately 50% was needed for the system to supply the required concentration of CO2 for an automatic (first shot) system discharge and that a tank volume of 92% was needed to provide the required amount of CO2 for a manual backup (second shot) discharge.

The licensee concluded that the CO2 system was operable because the tank volume at the time of discovery (75%) was sufficient to extinguish a fire on the initial discharge, and the NFPA code does not require double shot capability. Licensee practice was to refill the CO2 tank when level dropped below 70%; therefore, sufficient volume for a single discharge would be maintained unless a leak or equipment failure occurred. The team confirmed that, over the past two years, the CO2 tank volume had been maintained above the level required to attain the required CO2 concentration for an automatic system discharge. However, the team noted that the tank volume was above the level required for a second manual discharge less than 20% of the time over the two years before the deficiency was identified, and in one case the tank volume was below the required 92% for a period of 150 days.

The licensee immediately established a one hour fire watch for the cable tray mezzanine when it was determined that the system did not conform to the UFPARR and UFSAR specifications. The fire watch remained in effect until the tank was refilled above 92%. Subsequent corrective actions included revision of the tank volume limit in fire protection program administrative and surveillance procedures. The licensee also planned to evaluate the CO2 system design calculations to determine if changes were needed to tank level alarms or CO2 system timer settings.

CO2 Soak Time

During an assessment of Beaver Valley Unit 1 Appendix R issues, a licensee contractor questioned whether the initial test results for the cable tray mezzanine and other CO2 systems met design requirements. Results from the test of the CS-1 CO2 system, conducted on June 21, 1975, indicated that concentrations of greater than 52% were attained at elevations of 2 feet, 6 feet, and 7.5 feet above floor level. However, CO2 concentration was maintained above 52% for only 8 minutes, 3.5 minutes, and 1.5 minutes at the respective sample points. CR 01-0192 was initiated to evaluate the initial test data for required soak times.

For systems designed to suppress deep-seated fires, NFPA 12-1973 requires that the extinguishing concentration be maintained for a substantial period of time to assure complete extinguishment. NFPA 12-1973 does not specify a required soak time, but states that "the required extinguishment concentration shall be maintained for a sufficient period of time to allow the smoldering to be extinguished and the material to cool to a point at which re-ignition will not occur when the inert atmosphere is dissipated." A later version of NFPA-12 states that "after the design concentration is reached, the concentration shall be maintained for a substantial period of time, but not less than 20 minutes. Electrical cable fire suppression test results, documented in NUREG/CR-3656, "Evaluation of Suppression Methods for Electrical Cable Fires," indicated that a CO2 concentration of 50% was effective in extinguishing fully developed cable tray fires, provided there was sufficient soak time and the room was adequately sealed. Minimum soak times of 10 minutes and 15 minutes were identified for unqualified cables and IEEE-383 qualified cables, respectively.

Subsequent to the initial CO2 system testing in 1975, the licensee has upgraded fire penetration seals, modified dampers, and replaced fire door seals. Therefore, they expect that current CO2 retention times would be higher than the times observed in 1975. This expectation is also based on the consideration that it was common practice at the time of the initial testing to accept test results based on identification and correction of leakage paths without retesting. The licensee initiated a corrective action to have a vendor perform an analysis to determine expected CO2 concentrations over time with the enhanced fire area boundary integrity. The ability of the cable tray mezzanine CO2 system to perform its intended function of suppressing deep-seated fires is unresolved pending the results of the this analysis. **(URI 05000334/2001-005-02)**

.5 Post-Fire Safe Shutdown Emergency Lighting and Communications

a. Inspection Scope

The team observed the placement and aim of emergency light units (ELUs) throughout the plant to evaluate their adequacy for illuminating access and egress pathways and any equipment requiring local operation for post-fire safe shutdown. The team also evaluated installed and portable communication systems to determine if communications could be maintained in the event of a fire in the selected areas and during a shutdown from outside of the control room.

The team reviewed preventive maintenance procedures and surveillance procedures to determine if adequate surveillance testing was being accomplished to ensure operation of the emergency lights.

b. Findings

No findings of significance were identified.

.6 Safe Shutdown Capability

a. Inspection Scope

The team reviewed the Beaver Valley Unit 1 Updated Fire Protection Appendix ‘R’ Review to evaluate the methods and equipment used to achieve hot shutdown and cold shutdown, and to minimize the release of radioactivity following postulated fires in the selected risk significant fire areas. The team further reviewed piping and instrumentation drawings (P&IDs) for post-fire safe shutdown systems to determine required components for establishing flow paths, identify equipment required to isolate flow diversion paths, and verify appropriate components were on the safe shutdown equipment list. The team also performed field walk-downs to validate the equipment locations considered in the analysis and to evaluate the protection of the equipment from the effects of fires.

The team verified that the applicable requirements of 10 CFR 50, Appendix R, Sections III.G and III.L for achieving and maintaining safe shutdown were properly addressed. The team verified that systems necessary to assure the safe shutdown functions of reactivity control, reactor coolant makeup, reactor heat removal, and process monitoring were protected or independent from the selected areas. Where deviations from Appendix R requirements were identified, the team verified that applicable NRC exemptions had been approved and that conditions required by the exemptions were being properly implemented.

The team reviewed electrical drawings for the components controlled from the backup indicating panel (BIP) and other remote control circuits to ensure that proper isolation was provided for alternate shutdown capability for fires which required shutdown from outside of the control room. The team also reviewed BIP operability test procedures to determine if the licensee was appropriately testing the remote indication and control functions.

b. Findings

No findings of significance were identified.

.7 Operational Implementation of Safe Shutdown Capability

a. Inspection Scope

The team reviewed pre-fire plans for the selected areas to determine if appropriate information is provided to fire brigade members and plant operators to identify safe shutdown equipment and instrumentation, and to facilitate suppression of a fire that could impact safe shutdown. The team reviewed post-fire shutdown procedures (system and equipment recovery procedures, alarm response procedures, and abnormal procedures) for the selected areas to determine if appropriate information is provided to plant staff to perform required recovery actions to achieve and maintain safe shutdown.

The team walked down a postulated fire scenario which required operators to shutdown the plant from outside of the control room. The postulated fire was in the cable tray mezzanine (CS-1) and was assumed to quickly cause control room indications and controls to be unreliable, requiring plant shutdown from outside of the control room. Manual reactor trip from the control room was considered to be successful for the purposes of the scenario. All other actions were implemented from outside of the control room. Carbon dioxide gas migration (from automatic fire suppression system actuation in CS-1) was assumed, requiring operators to wear self-contained breathing equipment when realigning electrical equipment in the normal and emergency switchgear rooms. The postulated fire damaged electrical cables which caused a loss of offsite power (LOOP). The postulated fire also damaged control cables for all three river water (RW) pumps, requiring operators to diagnose the casualty and implement 1OM-56C.4.F-11, "RW and Emergency Generator Restoration," Rev. 1.

An operations crew, including a nuclear shift supervisor, a nuclear control operator, a nuclear operator, and a shift technical advisor performed a simultaneous walk-through of 1OM-56C, "Alternate Safe Shutdown From Outside the Control Room," and other procedures required to respond to the postulated fire scenario. The team observed each operator implement (or simulate) the applicable procedure sections and established a time line (Attachment 1) to validate whether time critical actions could be performed as required. The team evaluated the accessibility of the alternative shutdown operating stations and the accessibility of required manual action locations. The team also evaluated fire hazards in the vicinity of equipment requiring operator actions, and along the access and egress paths.

Additionally, the team conducted in-plant walkdowns of 1OM-56C.4.F-7, "Pneumatic Jumpering of Condenser Steam Dump Valves," Rev. 11 and 1OM-56C.4.F-9, "Cross-Connecting FPS Water Supply to the Neutron Shield Tank Cooler," Rev. 10 with fire protection engineers to verify procedure adequacy, tool and equipment availability, and the mechanical integrity of pneumatic jumper connections, necessary for plant cooldown.

The team evaluated whether minimum shift staffing was sufficient to implement 1OM-56C and other procedures required to achieve safe shutdown from outside of the control room. The team also reviewed training lesson plans and scenario guides for post-fire and alternative shutdown procedures, and discussed the procedures with licensed operators. Additionally, the team reviewed fire brigade and licensed operator qualifications to assure that personnel were qualified to perform required tasks for fire suppression and establishing safe shutdown conditions.

b. Findings

The team identified a deficiency in the procedure for shutdown from outside the control room which could have caused the protected EDG to be damaged, resulting in a loss of all AC power and loss of reactor coolant system makeup capability. The safety significance of this finding was very low because there was no impact on fire barriers, fire detection, or fire suppression capability, and the likelihood of the event is small. The failure to have adequate procedures to assure alternate shutdown capability with a loss of offsite power and loss of RW cooling was considered a non-cited violation of 10 CFR 50, Appendix R, Section III.L.

During the postulated fire scenario, a loss of all RW cooling occurred after the operators left the control room, followed shortly by a LOOP which caused both EDGs to automatically start and load. During the procedure walk-down, the protected train EDG (1-2 EDG) ran for approximately 15 minutes without cooling water (see Attachment 1) before an operator secured the EDG in accordance with 1OM-56C.4.D, “Nuclear Operator Procedure,” Rev. 16. This procedure did not direct the operator to verify adequate cooling to the protected EDG until after completing several other actions, including securing the unprotected EDG.

An engineering analysis determined that the EDG could run for 7 to 8 minutes without RW cooling, prior to EDG damage. As a result of this scenario, the 1-2 EDG would have been damaged and unrecoverable. This would in result in loss of all AC power, loss of reactor coolant system makeup capability, and interruption of reactor coolant pump (RCP) seal cooling.

Subsequent operator interviews and procedure walk-downs indicated that operators could detect the loss of RW cooling and secure both EDGs within several minutes upon arrival at the EDG building. Nuclear operators are trained to check for RW cooling to the EDGs by listening for flow in a storm drain outside of the entrance to the EDG building prior to entering the building. If no flow was detected, the operator could promptly take action to secure both EDGs.

The team concluded that 1OM-56C.4.D did not provide adequate direction to promptly verify RW cooling to the EDG, in order to assure that the protected EDG was not damaged. Although operators were trained to check for RW cooling prior to entering the EDG building, without procedural direction there is less assurance that the action will be performed, particularly under high stress conditions such as a shutdown from outside of the control room during a fire. The team also noted that the LOOP could occur earlier than it did in the postulated fire scenario and additional delays could be encountered before the operator was dispatched to the EDG building.

The team determined that the 1OM-56C.4.D procedure deficiency had a credible impact on safety, in that the delay in verifying adequate RW cooling to the running EDG could result in damage to the protected train EDG and a loss of all AC power. Without AC power, reactor coolant system makeup capability would be lost and safe shutdown could not be maintained. Using IMC-0609, "Significance Determination Process (SDP)," Appendix F, Fire Protection, Phase I, the team determined that the procedure deficiency was a degradation of a fire protection feature which could prevent operators from maintaining safe shutdown of the plant. The team further determined that the issue did not affect detection, manual or automatic suppression capability, or fire barriers. Therefore the issue screened out of the SDP process at step 1 of the Phase I Fire Protection SDP. The team determined that the procedure deficiency was a finding of very low safety significance (GREEN), because the likelihood of a fire which would cause a loss of all RW and necessitate shutdown from outside of the control room is small.

10 CFR 50, Appendix R, Section III.L.3, "Alternative and Dedicated Shutdown Capability," requires, in part, that alternative shutdown capability accommodate post-fire conditions where offsite power is not available and that procedures be in effect to implement this capability. Contrary to this requirement, 1OM-56C.4.D was inadequate to assure alternate shutdown capability with a loss of offsite power and loss of RW cooling. This violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy **(NCV 05000334/2001-005-03)**. The procedure deficiency is in the licensee's corrective action program as CR 01-3072.

.8 Safe Shutdown Circuit Analyses

a. Inspection Scope

The team reviewed the Beaver Valley Unit 1 Fire Protection Appendix 'R' Review to assess the adequacy of the methodology applied in the analysis for assuring that circuits required for safe shutdown were identified and protected. The team also reviewed the power and control cable routing for selected risk-significant post-fire safe shutdown components to determine if the cables were properly routed outside the fire areas of concern or protected against the effects of the postulated fires. The team reviewed selected plant electrical modifications that had been implemented to address control circuit fire vulnerabilities. The team also walked down certain portions of cable routing to confirm that the cables required for safe shutdown would not be impacted by the postulated fires.

The team reviewed electrical fuse and circuit breaker coordination studies to ensure that equipment needed for post-fire safe shutdown would not be impacted due to a lack of coordination. The team also reviewed testing, overhaul and preventive maintenance procedures for medium voltage circuit breakers to determine if the licensee was appropriately maintaining them in a state of readiness. These procedures were reviewed to determine if the circuit breakers that provide electrical power and provide protection to post-fire safe shutdown components could operate when called upon.

Due to the issuance of Change Notice 00-020 against Inspection Procedure 71111.05, "Fire Protection," the team did not review associated circuit issues during this inspection. This change notice has suspended this review pending completion of an industry initiative in this area.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

.1 Corrective Actions for Fire Protection Deficiencies

a. Inspection Scope

The team reviewed a self-assessment, an independent audit, and Quality Services surveillance reports conducted during the past two years. Selected condition reports (CRs) for fire protection and post-fire safe shutdown equipment were also reviewed. This review included the CRs initiated to address issues identified during the inspection. The team also reviewed a recent fire protection system health report, as well as selected outstanding and completed fire protection equipment work items. These reviews were conducted to determine if FirstEnergy was identifying fire protection program deficiencies and implementing appropriate corrective actions.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

.1 Exit Meeting Summary

The inspectors presented their preliminary inspection results to Mr. Saunders, Mr. Myers, and other members of the FirstEnergy staff at an exit meeting on June 8, 2001.

The inspectors asked whether any materials examined during the inspection should be considered proprietary. None of the information reviewed during the inspection was identified as proprietary.

(1) POSTULATED FIRE TIMELINE

T = 0	Fire reported in CS-1. Annunciators received in control room.
	Crew references 1OM-56B.3.B.2.
T = 6	Emergency switchgear ventilation secured IAW 1OM-56B.3.B.2.
	Crew references 1OM-56C. NSS decides to evacuate control room.
T = 12	NCO trips reactor IAW 1OM-56C.
	Operators leave control room.
	Fire causes loss of all river water (unbeknownst to crew).
T = 15	Operators assemble in Fire Brigade Room. Obtain copies of procedures.
T = 20	Operators leave Fire Brigade Room. Fire causes LOOP. Both EDGs start.
	NCO obtains biopack for entry into normal switchgear room.
	NSS verifies turbine tripped.
T = 26	NSS opens MSIV IA blowdown valves. TDAFWP starts.
	STA waiting in aux building for NSS to complete action.
T = 35	EDG #2 secured due to loss of river water.
T = 40	NSS directs implementation of 1OM-56C.4.F-11 to recover river water.
	NSS continues implementation of 1OM-56C.4.B.
T = 42	NO completes steps IV.B.2 - 4 of F-11.
	NSS “redirects” STA to perform Part A of STA attachment to F-11.
T = 55	NCO completes Part A of NCO attachment to F-11.
	NCO continues implementation of 1OM-56C.4.C.
	NSS commences BIP activation IAW 1OM-56C.4.F-1.
T = 58	STA completes Part A of STA attachment to F-11. (Note: 7 minutes deducted from timeline due to confusion on STA direction.)
T = 61	EDG #2 ready to load IAW F-11.
	NO recommences 1OM-56C.4.D.

(1) POSTULATED FIRE TIMELINE (Cont’d)

T = 65	Bus DF energized. RW pump B started.
T = 68	RW discharge valve full open. Charging pump 1B started.
T = 75	STA completes Part B of STA attachment to F-11.
	NO completes electrical switching in EDG room.
T = 80	NCO completes Part D of NCO attachment to F-11.
T = 81	BIP energized.
T = 82	STA recommences 1OM-56C.4.E at blender cubicle. Opens 1CH-28.
T = 85	NSS takes initial readings from BIP.
T = 94	NCO blocks open emergency switchgear room doors.
T = 99	NSS commences source range drawer hookup IAW F-1.
	NO arrives at BIP. Directed to control SG level.
T = 106	NCO arrives at BIP. Directed to operate ASDVs and HCV.
T = 110	Source range drawer energized.
T = 115	NSS completes SR drawer calibration check.
	STA arrives at BIP.
T = 120	Scenario complete.

(2) SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

FirstEnergy Nuclear Operating Company

J. Devine, Senior Nuclear Advisor
K. Frederick, Supervisor, Nuclear Engineering - Engineering Analysis & Assurance Section
W. Guntrum, Lead Nuclear Instructor
A. Hartner, Superintendent Nuclear Operations - Unit 1
I. Iannuzzi, Lead Nuclear Engineer
H. Kahl, Senior Nuclear Engineer
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J. Maracek, Supervisor, Regulatory Affairs
J. Miller, Senior Nuclear Engineer
L. Miller, Senior Nuclear Engineer
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R. Plummer, Associate Nuclear Analyst
R. Saunders, President, FirstEnergy Nuclear Operating Company

Nuclear Regulatory Commission

J. Linville, Chief, Electrical Engineering Branch

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000334/2001-005-03	NCV	Inadequate Procedure for Safe Shutdown from Outside of the Control Room Could Result in Damage to EDG. (Section 1R05.07)

Opened

05000334/2001-005-01	URI	Adequacy of Hemyc Cable Wrap Fire Barrier Test and Evaluation. (Section 1R05.02)
05000334/2001-005-02	URI	Capability of Cable Tray mezzanine CO2 System to Suppress Deep-Seated Fires. (Section 1R05.03)

Closed

None

Discussed

None

(2) SUPPLEMENTAL INFORMATION (Cont'd)

LIST OF ACRONYMS USED

ASDV	Atmospheric Steam Dump Valve
BIP	Backup Indicating Panel
CH	Charging
CO2	Carbon Dioxide
CRs	Condition Reports
CR-1	Control Room General Area
CR-4	Control Room Process Instrument Room
CS-1	Cable Tray Mezzanine
DCP	Design Change Package
EDG	Emergency Diesel Generator
ELU	Emergency Light Unit
FHA	Fire Hazards Analysis
HCV	Hand Control Valve
IA	Instrument Air
IEEE	Institute of Electrical and Electronics Engineers
IPEEE	Individual Plant Examination of External Events
LOOP	Loss of Offsite Power
MSIV	Main Steam Isolation Valve
NCO	Nuclear Control Operator
NCV	Non-Cited Violation
NFPA	National Fire Protection Association
NO	Nuclear Operator
NRC	Nuclear Regulatory Commission
NS-1	Normal Switchgear Room
NSS	Nuclear Shift Supervisor
P&IDs	Piping and Instrumentation Drawings
QA	Quality Assurance
PA-1E	Primary Auxiliary Building General Area E
RCP	Reactor Coolant Pump
RW	River Water
SCBA	Self-Contained Breathing Apparatus
SDP	Significance Determination Process
STA	Shift Technical Advisor
TDAFWP	Turbine Driven Auxiliary Feedwater Pump
UFPARR	Updated Fire Protection Appendix R Review
UFSAR	Updated Final Safety Analysis Report

(3) LIST OF DOCUMENTS REVIEWED

Piping and Instrumentation Drawings

8700-RM-406, “Valve Oper NO Diagram - Reactor Coolant System”
8700-RM-407, “Valve Oper NO Diagram - Chemical and Volume Control System”
8700-RM-410, “Valve Oper NO Diagram - Residual Heat Removal System”
8700-RM-411, “Valve Oper NO Diagram - Safety Injection System”
8700-RM-413, “Valve Oper NO Diagram - Containment Depressurization System”
8700-RM-414A, “Valve Oper NO Diagram - Sample System”
8700-RM-415, “Valve Oper NO Diagram - Component Cooling Water”
8700-RM-421, “Valve Oper NO Diagram - Main Steam”
8700-RM-422, “Valve Oper NO Diagram - Condensate”
8700-RM-424, “Valve Oper NO Diagram - Feedwater System”
8700-RM-430, “Valve Oper NO Diagram - River Water System”
8700-RM-431, “Valve Oper NO Diagram - Circulating Water System”
8700-RM-436, “Valve Oper NO Diagram - Emergency Diesel Generator Air Start System”
8700-RM-433-1, “Valve Oper NO Diagram - Fire Protection - Water”
8700-RM-433-2, “Valve Oper NO Diagram - Fire Protection - Water”
8700-RM-433-3, “Valve Oper NO Diagram - Fire Protection - CO2”

Control Circuit Schematics and Wiring Diagrams

8700-RE-1K, ”480V One-Line Diagram,” Sheet 4
8700-RE-1S, “480V One Line Diagram”
8700-RE-1AK, “120VAC One Line Diagram”
8700-RE-1AM, “480 V One-Line Diagram,” Sheet 14
8700-RE-1AR, “480V One Line Diagram”
8700-RE-3GK, “Wiring Diagram - BIP Transfer Switch Panel No. 1"
8700-RE-9GS, “Wiring Diagram - 480V MCC-1-E5 West CA Vault, Sheet 4"
8700-RE-10R, “Wiring Diagram- BIP - 125V DC Distribution”
8700-RE-11B, “Wiring Diagram - 120V AC Vital Bus 2 & 4"
8700-RE-11R, “Wiring Diagram - BIP - 120V AC Distribution”
8700-RE-18A, “Pressurized Heater Distribution Panels,” Sheet 1
8700-RE-21BU, Elementary Diagram Diesel Generator No. 1 Engine Controls
8700-RE-21BV. Elementary Diagram Diesel Generator No. 1 Engine Controls
8700-RE-21CA, Elementary Diagram Diesel Generator No. 2 Engine Controls
8700-RE-21BZ. Elementary Diagram Diesel Generator No. 2 Engine Controls
8700-RE-21CE, Elementary Diagram Diesel Gen 1& 2 Auto Loading Sequence
8700-RE-21DN, “Elementary Diagram - Reactor Coolant Vent System”
8700-RE-21FK, “Elementary Diagram - Component Cooling (CC)”
8700-RE-21FU, “Elementary Diagram - Charging and Volume Control (CH)”
8700-RE-2IHN, “Instrument Air,” Sheet 1
8700-RE-21JR, “Reactor Cooling, Pressurizer Heaters, Sheet 5"
8700-RE-21JS, “Reactor Cooling, Pressurizer Heaters, Sheet 6"
8700-RE-21JT, Elementary Diagram Reactor Cooling (Sheet 7 of 7)
8700-RE-21KW, Elementary Diagram River Water (Sheet 1 of 5)
8700-RE-22W, “Steam Generator System - Steam Generator No. 1 Narrow Range Level Protection I & II - Loop Diagram’s LT-FW474 & 475"
8700-RE-22X, “Steam Generator System - Steam Generator No. 2 Narrow Range Level Protection I & II - Loop Diagram’s LT-FW484 & 485"

(3) LIST OF DOCUMENTS REVIEWED (Cont.)

8700-RE-22Y, “Steam Generator System - Steam Generator No. 3 Narrow Range Level Protection I & II - Loop Diagram’s LT-FW494 & 495"
8700-RE-22BJ, “Reactor Coolant System - Pressurizer Level Protection Ch. II Level Control System - Loop Diagram LT-RC-460"
8700-RE-22BP, “Reactor Coolant System - Loop Isolation Valves Temp. Interlock Cold Leg Protection II - Loop Diag. TRB-RC-410, 420, 430"
8700-RE-22BM, “Reactor Coolant System - Wide and Narrow Range Pressure Loop Diagrams PT-RC-402, PT-RC-403, & TRB-RC-401"
8700-RE-25AR, “Outline 480V Motor Control Centers, Sheet 5"
8700-RE-25AQ, “Outline 480V Motor Control Centers, Sheet 4"
8700-RE-34AC, Cable Tray Designations - Control and Switchgear Area
8700-RE-36E, Wiring Diagram Penetration RCP-4E
8700-RE-100A, “4KV Station Service System,” Sheet 8

Engineering Evaluations/Modifications/Safety Evaluations/Change Requests

TER 7895, “Cable Mezzanine CO2 Pre-Discharge Alarm - Increase Warning Time to 60 Seconds”
TER 9715, “Retire BVPS-1 Control Room Area Smoke Detector Switches for HVAC System,” dated 10/31/95
TER 11561, “Penetration CR-735-259 Internal Conduit Seal Alternate Seal Arrangement and Temporary Conduit Seal Approval”
TER 12062, “East Cable Mezzanine Penetration ECM-725-213 Repair,” dated 8/17/98
TER 12743, “Portable Ventilation Requirements During a Fire in CR-2,” dated 9/9/99
TER 13386, “Administrative Revision to Unit 1 Updated Fire Protection Appendix R Review Report,” dated 8/10/00
TER 13719, Rev. 0, “Breaker Coordination Study for BV1 Appendix R Electrical Distribution System”
TER 13791, “Surveillance Limit for CO2 Storage Tank Level,” dated 5/25/01
EM 30494, “Inoperable Fire Barrier Between CR-3 and CR-4,” dated 10/16/85
DCP 1482, “Group 1 Fire Damper Replacement”
DCP 556, “Diesel Generator Circuit Modifications”
DCP 580, “Fire Wrap Ch-P-1B. Power Cable Raceways,” Rev. 1
DCP 667, “Fire Wrapping CH-P-1B Power Cable at 735 PAB,” Rev. 0
DCP 689, “Appendix R Modification of Diesel Generator Building CO2 Circuit”

Calculations

8700-DEC-0187, Rev. 0, “Hemyc Blanket Fire Barrier Configurations”
8700-DMC-2975, Addendum 1, “BV1 ESGR Area Heat-up Following Recovery of Loss of All AC”
8700-DMC-3467, Rev. 1, “BVPS Common Control Room Area Heatup Rates Following a Loss of Unit 1 HVAC”
8700-DMC-3507, Rev. 0, “Emergency Diesel Generator Room 1-2 Heat up with Appendix R Scenario - Loss of Ventilation”
8700-DMC-3509, Rev. 0, “HHSI Pump Oil Temperature Following Loss of River Water”
10080-DMC-3443, “GL 86-10 Evaluation of Excessive Clearances Between Fire Rated Doors and Frames”
B-81, “Heat Gain Analysis to Emergency Switchgear Room (1AE & 1DF) During a Complete

Loss of Ventilation (First Hour)”

Procedures

1OM-33.4.AAN, “Cable Tray Mezzanine Fire”
1OM-33.4.ADF, “A/C Equipment or Cable Sprdg Rm Fire”
2OM-44A.4.E, “Purging Smoke or Toxic Gases from the Control Room”
1OM-44A, “Area Ventilation Systems - Control Area”
1OM-53.A.1.2-B(ISS1C), “Establishing RCP CCR Cooling and Seal Injection”
1OM-53.A.1.ECA-0.0(ISS1C), “Loss of All Emergency KV AC Power”
1OM-56B, “Fire Prevention and Control” (Pre-Fire Plan Strategies and Recovery Procedures)
1OM-56C, “Alternate Safe Shutdown From Outside Control Room”
AOP 1.33.1A, “Control Room Inaccessibility”
NPDAP 3.5, “Fire Protection”
NEAP-2.13, “Technical Evaluation Reports”
NOP-LP-2001, “Condition Report Process”
1/2ADM-2004, “Design Interface Reviews and Evaluations”
1/2ADM-2007, “Engineering Change Packages”
1BVT 01.33.05, “Fire Rated Assemblies and Electrical Penetration Flood Seals Visual Inspection”
1OST-33.7, “Motor Driven Fire Pump Operation Test”
1OST-33.13B, “Deluge Valve Fire Protection System Instrument Test”
1OST-33.13C, “Ten Ton CO2 Fire Protection System Test”
1OST-33.16, “Smoke Detector Instrumentation Test”
1/2PMP-33FP-Fire Doors-1M, “Periodic Inspection of Fire Doors”
1/2PMP-75VS-VNT-4M, “Ventilation System Fire Damper Maintenance and Trip Check”
1/2PMP-E-36-015, “ITE Medium Voltage Circuit Breaker Inspection and Test Model 5HK-250/350"
1PMP-E-37-011, “General Electric Low Voltage Circuit Breaker Inspection and Test Model AK-3A & 7A-25"
1PMP-E-37-012, “General Electric Low Voltage Circuit Breaker Inspection and Test Model AK-3A-50S and AKS-7A-50"
1PMP-37-SS-Linestarter-2E, “Linestarter Inspection”
1PMP-38VB-UPS-1-3I, “Uninterruptible Power Supply (Solid State Controls) No. 1 Maintenance”
1PMP-39DC-BKR-1E, “Battery Air Circuit Breaker Inspection, GE AK-2A-25"
1/2RCP-38A-PC, “Calibration of ITE/ABB Single Phase Overcurrent Relays ITE Type 50 and ITE Type 51 (With SCR Outputs)”
1/2RCP-38B-PC, “Calibration of ITE/ABB Three Phase Overcurrent Relays Type 51 With SCR Outputs”
1RCP-7-PC, “Calibration of Westinghouse/ABB Overcurrent Relays, Type COM-5"
1RCP-8A-PC, “Calibration of Westinghouse/ABB Overcurrent Relays Type CO”
1LCP-56-BIP, Rev. 2, Loop Calibration Procedure BIP Instrumentation Calibration
1OST-45.9, Rev. 5, BIP Instrumentation and Source Range Indication Test
1OST-45.10, Rev. 1, BIP Valve Control Switch Test
1LCP-56-BIP, Rev. 2, Check/Calibration of the Back-Up Indicating Panel Instrumentation

Training Documents

Training Lesson Plan NLRT-1.56C, “Non-Licensed Operator Training”

(3) LIST OF DOCUMENTS REVIEWED (Cont.)

Training Lesson Plan LP-SQS-56C, “Alternate Safe Shutdown From Outside Control Room”
Training Lesson Plan LP-STA-71B, “Alternate Safe Shutdown In-Plant Tour”
Training Lesson Plan LP-STA-71A, “Alternate Safe shutdown From Outside Control Room”
Training Lesson Plan LP-SQS-56C.2, “OM56C- Appendix ‘R’ In-Plant Training
Training Lesson Plan LP-LRT-VII-66, “Alternate Safe Shutdown”

<u>Audits, Assessments, and Corrective Action Program Documents</u>

Fire Protection Self-Assessment, dated May 27, 1999
ORC Fire Protection Audit (BV-C-99-07), dated July 16, 1999
Quality Services Surveillance No. 3-MIS-57-99, Fire Brigade Drill 11/17/99
CNRB Fire Protection Audit (BV-C-00-07), dated July 11, 2000
Unit 1 Drill Report - 10M56C Procedure, “Post-Fire Alternate Safe Shutdown From Outside Control Room,” dated March 30, 2001
1-MIS-54-00, dated 4/24/00
3-MIS-93-00, dated 11/14/00
1-MIS-40-99, dated 4/22/99
1-MIS-41-99, dated 5/20/99
3-MIS-43-99, dated 5/21/99
3-MIS-44-99, dated 5/26/99
CR 99-0895
CR 99-1171
CR 99-1340
CR 99-1440
CR 99-1969
CR 99-3385
CR 99-3395
CR 00-1553
CR 00-1900
CR 00-1969
CR 01-0108
CR 01-0145
CR 01-0198
CR 01-0251
CR 01-0264
CR 01-1102
CR 01-1546
CR 01-2163
CR 01-2289
CR 01-2371
CR 01-2577
CR 01-2835
CR 01-2996*
CR 01-3009*
CR 01-3023*
CR 01-3032*
CR 01-3060*
CR 01-3062*
CR 01-3063*
CR 01-3064*

(3) LIST OF DOCUMENTS REVIEWED (Cont.)

CR 01-3065*
CR 01-3072*
CR 01-3075*
CR 01-3082*
CR 01-3090*
CR 01-3305*
CR 01-3325*
CR 01-3329*
CR 01-3342*
CR 01-3375*

* Denotes CR initiated during inspection.

Miscellaneous Documents

Updated Fire Protection Appendix R Review, Revision 16
- Section 1 - Introduction
- Section 2 - Historical Background
- Section 3 - Unit Description
- Section 4 - Shutdown Capability Summary
- Section 5 - Electrical Analysis
- Section 6 - Resolution of Problem Areas
- Section 8 - ID of High/Low Pressure System Interfaces
- Section 9 - Updated Response to NRC Staff's Generic Letter
- Section 10 - Appendix R Requirements J and O
- Section 11 - Exemptions and Deviations
- Section 12 - Schedule for Compliance

Beaver Valley Unit 1 Individual Plant Examination of External Events (IPEEE)
- Chapter 1 - Executive Summary
- Chapter 4 - Internal Fire Analysis
- Chapter 6 - Licensee Participation and Internal Review Team
- Chapter 7 - Plant Improvements and Unique Safety Features
- Chapter 8 - Summary and Conclusions
- Appendix E.1 - Location Characteristics Table (selected areas)
- Appendix E.3 - Propagation Pathway Credibility Assessment
- Appendix F.1 - Fire Ignition Frequency Assessment and Results
- Appendix F.2 - Fire Frequency Apportionment
- Appendix F.3 - Location Scenario Impact Estimation
- Appendix G - Detailed Fire Scenario Calculations (selected areas)

Fire Submittal Screening Review Technical Evaluation Report: Beaver Valley Unit 1, Rev. 2
WOG 2000 Reactor Coolant Pump Seal Leakage Model for Westinghouse Pressurized Water Reactors (WCAP 15003, Rev. 0)
Low Pressure CO2 System Field Test Report, dated 6/21/75
Specification for Special Fire Extinguishing Systems for Low Pressure Carbon Dioxide Fire Protection for Beaver Valley Power Station Duquesne Light Company, dated 4/1/74
System Health Report BVPS Unit 1 Fire Protection System #33, 1st Quarter 2001
OMDR 1-90-0232 and Letter ND1NSM:4285, dated 11/28/89, regarding Information Notice 89-52, "Potential Fire Damper Operational Problems"