September 20, 2000

Mr. Robert M. Bellamy
Site Vice President
Entergy Nuclear Generation Company
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360 - 5599

SUBJECT: PILGRIM - NRC FIRE PROTECTION INSPECTION REPORT NO. 05000293/2000-004

Dear Mr. Bellamy:

This letter forwards the results of a triennial fire protection team inspection conducted on August 14-18, 2000, at Pilgrim Nuclear Power Station. The preliminary results of the inspection were discussed with Mr. T. Sullivan, Vice President of Operations, and other members of your staff at an exit meeting on August 18, 2000. Updates to the preliminary results were discussed in a telephone call on August 28, 2000, between Mr. W. Lobo of your licensing group and Mr. R. Fuhrmeister of Region I.

This inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The purpose of the inspection was to evaluate your post-fire safe shutdown capability and fire protection program. Within these areas, the inspection consisted of selected examinations of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC identified two issues that were evaluated under the risk significance determination process and were determined to be of very low safety significance (Green). These issues have been entered into your corrective action program and are discussed in the summary of findings and the body of the attached inspection report. These issues were determined to involve violations of NRC requirements, but because of their very low safety significance the violations are not being cited. If you contest these non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Pilgrim facility.

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Sincerely,

/RA/

Wayne D. Lanning, Director
Division of Reactor Safety

Docket Nos. 05000293
License Nos. DPR-35

Enclosure: NRC Inspection Report 05000293/2000-004

cc w/encl:
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J. Alexander, Director, Nuclear Assessment Group
D. Tarantino, Nuclear Information Manager
S. Brennion, Regulatory Affairs Department Manager
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R. Hallisey, Department of Public Health, Commonwealth of Massachusetts
The Honorable Therese Murray
The Honorable Vincent DiMacedo
Chairman, Plymouth Board of Selectmen
Chairman, Duxbury Board of Selectmen
Chairman, Nuclear Matters Committee
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	05000293
License Nos:	DPR-35
Report No:	05000293/2000-004
Licensee:	Entergy Nuclear Generation Company
Facility:	Pilgrim Nuclear Power Station
Location:	600 Rocky Hill Road Plymouth, MA 02360
Dates:	August 14 - 18, 2000
Inspectors:	R. Fuhrmeister, Sr. Reactor Inspector, Division of Reactor Safety (DRS) L. Scholl, Senior Reactor Inspector, DRS T. Walker, Senior Reactor Inspector, DRS C. Cahill, Reactor Inspector, DRS L. James, Reactor Inspector, DRS K. Sullivan, Contract Engineering Support
Approved By:	William H. Ruland, Chief Electrical Engineering Branch Division of Reactor Safety

Table of Contents

SUMMARY OF FINDINGS

IR 05000293/2000-004, on 08/14/00 - 08/18/00, Entergy Nuclear Generation Company, Pilgrim Nuclear Generating Station, Fire Protection

This inspection was conducted by a team of regional specialists with support from a national laboratory contractor. This inspection identified two green issues, which were both non-cited violations. The significance of issues is indicated by their color (green, white, yellow, red) and was determined using the Significance Determination Process.

Cornerstone: Mitigating Systems

- Green. The NRC identified that emergency lighting units (ELUs) were not installed to support manual operation of the service water outlet valves for the reactor building closed cooling water heat exchangers. Additionally, these valves were not accessible for local, manual operation. Local, manual operation of these valves would be required in certain circumstances for post-fire shutdown. This finding is characterized as a condition of very low safety significance in accordance with the Fire Protection Significance Determination Process because it does not affect fire barriers, or fire detection or suppression capability. (Sections 1R05.3 and 1R05.9)

- Green. Emergency diesel generator (EDG) watt-meter cables in the cable spreading room, which could be damaged by a fire, were neither protected nor isolated as part of the Appendix R modifications. This led to the potential for a cable spreading room fire to cause a loss of the EDGs on the start of a residual heat removal pump, resulting in a station blackout condition in the post-fire operating environment. The Significance Determination Process characterizes this finding as a condition of very low safety significance because of the ability to diagnose the problem and recover electrical power. (Section 4OA2.2)

Report Details

Summary of Plant Status:

The unit operated at or near 100% power for the duration of the inspection.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems

1R05 Fire Protection

.1 Fire Barrier Penetration Seals

a. Inspection Scope

During plant tours, the team randomly selected three fire barrier penetration seals for detailed inspection to verify proper installation and qualification. The team reviewed associated design drawings, Promatec qualification test reports, Pitt-Char modifications of type III-T seals (specification M-570), and fire test CTP 1147. The team compared the observed in-situ seal configurations to the design drawings and tested configurations. The team also compared the penetration seal ratings with the ratings of the barriers in which they were installed.

b. Issues and Findings

There were no findings identified.

.2 Raceway Fire Barrier Systems

a. Inspection Scope

The team reviewed the adequacy of the design of the fire area boundaries, raceway fire barriers, fire doors and fire dampers for the plant areas selected.

b. Issues and Findings

There were no findings identified.

.3 Post-Fire Safe Shutdown Emergency Lighting

a. Inspection Scope

The team observed the placement and aim of emergency lighting units (ELUs) throughout the selected fire areas to evaluate their adequacy for illuminating access and egress pathways and any equipment requiring local operation for post-fire shutdown.

b. Issues and Findings

The team walked down the manual operator actions specified in procedure 2.4.143.1, “Shutdown with a Fire in the Reactor Building East (Fire Area 1.9)” and 2.4.143.2, “Shutdown with a Fire in the Reactor Building West (Fire Area 1.10)” and identified that ELU’s were not provided to support the manual operation of valves MO3800 and MO3805 (service water outlet valves for the reactor building closed cooling water heat exchangers). 10 CFR 50, Appendix R, Section J, requires that emergency lighting units with at least an 8-hour battery powered supply be provided in all areas needed for operation of safe shutdown equipment. The failure to provide ELU’s for the manual operation of valves MO3800, and MO3805 is a violation of this requirement. This issue has been entered into Entergy’s corrective action program as problem report 00.9324. The significance determination process characterized this condition as being of very low risk significance because it does not affect fire barriers, or fire detection or suppression capabilities. This violation of 10 CFR 50, Appendix R, Section J, is being treated as a Non-Cited Violation **(NCV 050000293/2000-004-01)**, consistent with Section VI.A. of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368).

.4 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 the control of ignition sources. The team also reviewed the implementation and results of several Fire Hazard(s) Inspections conducted in accordance with procedure number 8.B.20. These reviews were performed to evaluate the implementation of fire protection program administrative controls.

b. Issues and Findings

There were no findings identified.

.5 Manual Fire Suppression Equipment

a. Inspection Scope

The team walked down selected standpipe systems, CO_2 hose reels, and portable extinguishers. The team also inspected the fire brigade’s protective ensembles for structural fire fighting and portable communications equipment.

b. Issues and Findings

There were no findings identified.

.6 Alternative Shutdown Capability

a. Inspection Scope

The team reviewed the Pilgrim Nuclear Power Station (PNPS) Appendix R Safe Shutdown Analysis Report (Calculation PS-32, Revision 4), Procedure No. 2.4.143.1, Shutdown with a Fire in Reactor Building East (Fire Area 1.9), Revision 8, and Procedure No. 2.4.143.2, Shutdown with a Fire in Reactor Building West (Fire Area 1.10), Revision 8, to evaluate the methods and equipment used to achieve hot shutdown following postulated fires in the B Switchgear Room (Fire Zone 2.1), B Reactor Building Closed Cooling Water (RBCCW) Room (Fire Zone 1.22), and the Vital Motor Generator Set Room (Fire Zone 3.5) at the Pilgrim Nuclear Power Station. The team further reviewed piping and instrumentation drawings for post-fire shutdown systems to determine required components for establishing flow paths, identify equipment required to isolate flow diversion paths, and verify appropriate components are on the safe shutdown equipment list. The team also performed field walkdowns to evaluate the protection of the equipment from the effects of fires.

b. Observations and Findings

There were no findings identified.

.9 Operational Implementation of Alternative Shutdown Capability

a. Inspection Scope

The team reviewed post-fire shutdown procedures for the B Switchgear Room (Fire Zone 2.1), the B Reactor Building Closed Cooling Water (RBCCW) room (Fire Zone 1.22), and the Vital Motor Generator Set Room (Fire Zone 3.5) to determine if appropriate information is provided to plant operators to identify protected equipment and instrumentation and if recovery actions specified in post-fire shutdown procedures consider manpower needs for performing restorations and area accessibility. The team also reviewed training lesson plans and job performance measures for the alternative shutdown procedures, discussed training with licensed operators, reviewed the Appendix R locker surveillance, reviewed select alternate shutdown panel tests, reviewed minimum shift manning required by technical specifications, and evaluated the accessibility of the alternative shutdown operating stations and the accessibility of required manual action locations.

b. Observations and Findings

In accordance with the post-fire shutdown procedures, manual actions would be required to open MO3800 and MO3805 (service water outlet valves for the reactor building closed cooling water heat exchangers) to achieve hot shutdown for a fire in certain areas of the reactor building. These valves are located approximately 20 feet above the floor elevation. No ladders or platforms were provided for access to and operation of these valves. This issue was entered into Entergy's corrective action program as problem report 00.9324. This issue is related to the lack of emergency

lighting discussed in Section 1RO5.3 of this report, and is considered part of the same finding.

.10 Fixed Fire Suppression Systems

a. Inspection Scope

The team reviewed the adequacy of the automatic Halon fire suppression systems for the cable spreading room. This included a walkdown of the system and a review of the discharge and functional tests.

b. Issues and Findings

There were no findings identified.

.11 Safe Shutdown Circuit Analyses

a. Inspection Scope

The team reviewed calculation PS-32, “Appendix R Safe Shutdown Analysis Report,” to assess the adequacy of the methodology applied in the licensee’s analysis. The team also reviewed assumptions utilized in the analysis, the application of NRC guidance and the adequacy of engineering evaluations of design vulnerabilities such as the absence of full protective device coordination in some areas.

The team reviewed cable routing for a sample of components required for post-fire safe shutdown to determine if the cables were properly routed outside the fire area of concern or protected against the effects of fire. The team also reviewed selected circuit schematic drawings to determine if the licensee had properly evaluated the circuits for the effects of hot shorts, open circuits and shorts to ground.

The team reviewed selected electrical power buses for various voltage levels to verify that electrical protective device coordination existed for equipment needed to conduct post-fire safe shutdown activities. The adequacy of the licensee’s evaluation of a bus that lacked full coordination was also reviewed.

The team reviewed report 86XE-2CR-Q-E1, “Report on High Impedance Faults for 10CFR50 Appendix R,” dated December 19, 1986, to confirm that the potential effects of multiple high impedance faults had been evaluated and that the necessary actions had been incorporated into the affected procedures.

The team reviewed the electrical isolation capability of selected equipment needed for post-fire safe shutdown to ensure that such equipment could be operated locally if needed.

The team reviewed the licensee’s evaluation and actions taken in response to NRC Information Notice 92-18, “Potential for Loss of Remote Shutdown Capability During a Control Room Fire.”

The team reviewed the routing of the power and control circuit cables for the main seal oil pump and the emergency seal oil pump to determine if a fire in the “B” switchgear and load center room could cause a loss of seal oil for the main generator, resulting in a second fire.

b. Issues and Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

.1 Corrective Actions for Fire Protection Deficiencies

a. Inspection Scope

The team reviewed the Fire Impairments Log, a list of corrective maintenance action requests against post-fire safe shutdown equipment, and selected action requests for post-fire safe shutdown equipment to evaluate the effectiveness of corrective actions and the prioritization for resolving fire protection related deficiencies. The team also reviewed recent Quality Assurance Surveillance and Audit Reports, and Engineering Self-Assessments of the fire protection program to determine if Entergy was identifying program deficiencies and implementing appropriate corrective actions.

b. Issues and Findings

No findings were identified.

.2 (Closed) LER 50-293/1999-011. Postulated Fire in Cable Spreading Room Potentially Affecting Safe Shutdown

This LER identified a condition outside the design basis of the plant. Specifically, emergency diesel generator (EDG) watt-meter cables in the cable spreading room, which could be damaged by a fire, were neither protected nor isolated as part of the Appendix R modifications. This led to the potential for a cable spreading room fire to cause a loss of the EDGs on the start of a residual heat removal pump, resulting in a station blackout condition in the post-fire operating environment. This condition was resolved under problem report 99.9549, by a revision to Procedure 2.4.143, “Shutdown from Outside Control Room.” The revision added steps to unplug the voltage-controlled overcurrent relays at the switchgear cubicles for the EDG output breakers.

The team reviewed routing of control cables for the offsite supplies to 4.16 kV busses A5 and A6, and compared them to the routing of the watt-meter cables for the diesel generators, and control power cables for the reactor core isolation cooling system (RCIC) and high pressure coolant injection system (HPCI) discharge valves. The team concluded that a fire in 480VAC load center B6 could potentially damage the control

power cables for all of the power supplies to the A5 and A6 busses as well as the HPCI and RCIC discharge valves.

The risk significance of this issue was evaluated by a Region I Senior Reactor Analyst (SRA). Operation of either the HPCI or RCIC systems is possible with this fire configuration through Pilgrim Procedure No. 2.4.143, Shutdown from Outside Control Room. However, the recovery of a.c. power would depend on operator actions. Using the fire ignition frequency for the cable spreading room in the Pilgrim Nuclear Power Station Independent Evaluation for External Events, and crediting the fire brigade and cable spreading room gaseous fire suppression system as effective, a Phase 3 SDP was performed. The NRC's standardized plant analysis risk (SPAR) model human error worksheets were used to calculate the probability of error in both the diagnosis phase and the action portions of the a.c. power recovery task. A relatively low human error probability resulted because of both the number of on-site and off-site sources available for recovery and also because the time for recovery, 12 hours, is relatively long. This evaluation determined that the condition had a very low risk significance.

The failure to isolate all potentially affected cables within the cable spreading room constitutes a violation of the requirement of 10 CFR 50, Appendix R, Section III.G.3 that the alternative shutdown capability be independent of the area of concern. This violation of 10 CFR 50, Appendix R, Section III.G.3, is being treated as a Non-Cited Violation **(NCV 050000293/2000-004-02)**, consistent with Section VI.A. of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368).

4OA6 Meetings

.1 Exit Meeting Summary

The inspectors presented their preliminary inspection results to Mr. T. Sullivan and other members of the Entergy staff at an exit meeting on August 18, 2000.

Updates to the preliminary results were discussed in a telephone call on August 28, 2000, between Mr. W. Lobo of Entergy's licensing group, and Mr. R. Fuhrmeister of Region I.

The inspectors asked whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified during the course of the inspection.

PARTIAL LIST OF PERSONS CONTACTED

Entergy
T. Sullivan, Vice-President of Operations
H. Oheim, Director of Design Engineering
S. McAllister, Superintendent of Maintenance Programs
V. Fallacara, Operations Manager
B. Lyons, Operations Support Supervisor
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E. Olson, Assistant Operations Manager
J. Bonner, Supervisor of Electrical Design Engineering
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C. McMorrow, Sr. Fire Protection Engineer
P. Sullivan, Sr. Engineer, Assessment Group
F. DiCristofaro, Sr. Operations Engineer
F. Famular, Sr. Engineer, Maintenance Programs
R. Levin, Electrical Design Engineer
W. Lobo, Licensing Engineer
F. McGinnis, Systems and Safety Assessment Engineer

Nuclear Regulatory Commission
R. Laura, Senior Resident Inspector
R. Arrighi, Resident Inspector

Brookhaven National Laboratory

K. Sullivan

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

NCV 050000293/2000-004-01	Failure to install emergency lighting units for manual operation of heat exchanger service water outlet valves
NCV 050000293/2000-004-02	Failure to ensure that the alternate shutdown capability for the cable spreading room was electrically and physically independent of the area

Closed

LER 050000293/1999-011	Postulated Fire in Cable Spreading Room Potentially Affecting Safe Shutdown

Discussed

None

LIST OF ACRONYMS USED

AC	Alternating Current
ELU	Emergency Lighting Unit
EOP	Emergency Operating Procedure
JPM	Job Performance Measure
P&ID	Piping and Instrumentation Drawing
PNPS	Pilgrim Nuclear Power Station
RBCCW	Reactor Building Closed Cooling Water
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
SE	Safety Evaluation
SSW	Salt Service Water

LIST OF DOCUMENTS REVIEWED

Piping and Instrumentation Drawings

M212 Sheet 1, P&ID Service Water System, Revision E77
M215 Sheet 1, P&ID Cooling Water System Reactor Building, Revision E48
M215 Sheet 2, P&ID Cooling Water System Reactor Building, Revision E46
M215 Sheet 3, P&ID Cooling Water System Reactor Building, Revision E37
M241 Sheet 1, P&ID Residual Heat Removal System, Revision E71
M241 Sheet 2, P&ID Residual Heat Removal System, Revision E41
M242, P&ID Core Spray System, Revision E45
M252 Sheet 1, P&ID Nuclear Boiler, Revision E56
M252 Sheet 2, P&ID Nuclear Boiler, Revision E50

Control Circuit Schematics / Wiring Drawings

E7-15-10, Rev. E1, Miscellaneous Circuits Substation 1 Load Center
E27, SH 1, Rev. E23, Diesel Generator "1" X107A
E27, SH 2, Rev. E21, Diesel Generator "2" X107B
E170, Rev. E10, Salt Water Service System
E171, Rev. E4, Salt Water Service Water System
E176, SH 1, Rev. E8, Reactor Building Closed Cooling Water System
E176, SH 2, Rev. E8, Reactor Building Closed Cooling Water System
E177, Rev. E8, Closed Cooling Water System - Reactor Building
E532, SH 1 Rev. E3, Diesel Generator Isolation Panels C160 & C161
E532, SH 2 Rev. E0, Diesel Generator Isolation Panels C160 & C161
E534, Rev. E3, 125V DC Auto Transfer Schemes
E5000, Rev. E10, Residual Heat Removal System Motor Operated Valves
E5002, Rev. E13, Primary Containment Isolation System
E5009, Rev. E16, Containment Spray Motor Operated Valves
E5011, Rev. E6, Reactor Shutdown Cooling System Isolation Motor Operated Valve
E72, Rev. E4, Turbine Generator Hydrogen Seal Oil Pumps

Calculations

Calculation Comment Sheet 86XE-2ER-0-E2, Report on High Impedance Faults for 10 CFR 50 Appendix R	
M794, Rev. 0	MO 1001-29B and MO 1001-43B Locked Rotor "Hot Short" Evaluation
PS-63, Rev. 0	Bus B1, B2, &B6 Breaker Settings 480 Volt Switchgear
PS-31, Rev. 1	D-C System Overcurrent Protection Coordination Study - Comment Sheet No. PS31-5

Procedures

EOP-01	Reactor Pressure Vessel (RPV) Control, Revision 5
EOP-17	Alternative RPV Depressurization, Revision 2
Procedure No. 2.1.6	Reactor Scram, Revision 45
Procedure No. 2.1.26	Inventory of Alternate Shutdown and EOP Support Tools, Revision 12

Procedure No. 2.2.19.5	Residual Heat Removal Modes of Operation for Transients, Revision 6
Procedure No. 2.2.20	Core Spray, Revision 48
Procedure No. 2.2.23	Automatic Depressurization System, Revision 25
Procedure No. 2.2.30	Reactor Building Closed Cooling Water (RBCCW) System, Revision 43
Procedure No. 2.2.32	Salt Service Water System (SSW), Revision 57
Procedure No. 2.4.143	Shutdown From Outside Control Room, Revision 20
Procedure No. 2.4.143.1	Shutdown with a Fire in Reactor Building East (Fire Area 1.9), Rev. 8
Procedure No. 2.4.143.2	Shutdown with a Fire in Reactor Building West (Fire Area 1.10), Rev. 8
Procedure No. 5.3.19	Loss of 120V AC Safeguard Buses Y4 and Y41, Revision 16
Procedure No. 8.5.15	Core Spray Motor Operated Valve Operability From Alternate Shutdown Panel, Revision 14
Procedure No. 8.5.3.8	RBCCW Pump and Valve Alternate Shutdown Panel Test, Revision 9
Procedure No. 8.9.13	Diesel Generator Alternate Shutdown Panel Test, Revision 12
Procedure No. 1.4.3	Combustible Controls for Pilgrim Station
Procedure No. 1.5.5	Hotwork Fire Safety
Procedure No. 2.2.29	Smoke and Heat Detection Systems
Procedure No. 2.4.54	Loss of All Fire Suppression Pumps or Loss of Redundancy in the Fire Water Supply System
Procedure No. 2.4.143	Shutdown from Outside Control Room
Procedure No. 2.4.143.1	Shutdown with a Fire in the Reactor Building East (Fire Area 1.9)
Procedure No. 2.4.143.2	Shutdown with a Fire in the Reactor Building West (Fire Area 1.10)
Procedure No. 3.M.4-72	Diesel Fire Pump Engine Maintenance
Procedure No. 5.5.1	General Fire Procedure
Procedure No.5.5.2	Special Fire Procedure
Procedure No. 8.B.15	Functional and full Design Test of Fire Pumps, completed 8/24/95
Procedure No. 8.B.15	Functional and full Design Test of Fire Pumps, completed 5/23/98
Procedure No. 8.B.15	Functional and full Design Test of Fire Pumps, completed 7/18/98
Procedure No. 8.B.15	Functional and full Design Test of Fire Pumps, completed 9/8/99
Procedure No. 8.B.20	Fire Hazard(s) Inspection
Procedure No. 8.B.29	Inspection of Fire Barriers
Procedure 2.4.157	Loss of Hydrogen Seal Oil

Training Materials

Instructional Module, O-RQ-03-02-02	Special Fire Events, Revision 2
Instructional Module, O-RQ-02-02-02	Shutdown Outside the Control Room, Revision 1
Job Performance Measure (JPM) 205-03	Place RHR in Torus Cooling From Outside the Control Room, Revision 6
JPM-205-12	Alternate power to RHR Valves, Revision 1
JPM-218-01	Safety Relief Valve Operation from Outside the Control Room, Revision 5
JPM-239-03	MSIV Closure from Outside the Control Room, Rev. 0
JPM-262-05	Operate A 480 VAC Breaker Locally, Revision 5

JPM-262-03	Local Operation of 4160 VAC Breakers During Shutdown from Outside the Control Room, Revision 3
JPM-264-04	Emergency Diesel Generator Operation Outside the Control Room, Revision 7

Corrective Action Program Documents

PR 00.9320	Fire Barrier 209.1A Degraded
PR 00.9324	Missing Appendix R Emergency Lighting Units
PR 00.9325	Fire Barrier 194-505 Degraded
PR 99.9549	Apparent Appendix R issue affecting ability of the EDGs to load following a fire in the cable spreading room
PR 99.9549.00	Apparent Appendix R issue affecting ability of the EDGs to load following a fire in the cable spreading room, Root Cause Analysis
PR 99.9549.01	Apparent Appendix R issue affecting ability of the EDGs to load following a fire in the cable spreading room, Reportability Review
PR 99.9549.02	Apparent Appendix R issue affecting ability of the EDGs to load following a fire in the cable spreading room, Engineering Evaluation
PR 99.9549.03	Apparent Appendix R issue affecting ability of the EDGs to load following a fire in the cable spreading room, Action Assignment

Drawings

A317	Reactor & Turbine Building Floor Plan at El. 23'-0" Fire Barrier System
C120	Removable Wall Panel Upgrade to the Three (3) Hour Firewall Switchgear Room B
M277-13	Pitt-Char Modification Fire Seal
E1, Rev. E17	Single Line Diagram - Station
E7, Rev. E7	Single Line Meter and Relay Diagram - 4160 Volt System
E8, Rev. E19	Single Line Meter and Relay Diagram - 4160 Volt Breaker A409, 480 Volt Load Center B8, 480 Volt Motor Control Center B30, B17A & B18A
E9, Rev. E49	Single Line Meter and Relay Diagram 480 Volt System - Load Centers & Motor Control Centers B10 & B20
E10, Rev. E36	Single Line Diagram - 480V System Motor Control Centers B14, B15, B17, B18, B28 & B29
E11, Rev. E30	Single Line Diagram - 480V System Motor Control Centers B13, B22, B23, B25 & B26
E12, Rev. E30	Single Line Diagram - 480V System Motor Control Centers B16, B19, B21 & B24
E13, Rev. E71	Single Line Relay & Meter Diagram 125V & 250V DC Systems
E14, Rev. E30	Single Line Diagram 120V Instrument AC, Vital & Reactor Protection AC Systems & 24VDC Power System
M12, Rev. E15	Equipment Locations, Turbine Building Plan Ground Floor EL 23'-0"
M15, Rev. E20	Equipment Locations, Reactor Building Plan Basement EL (-) 17'6"
E320, Rev. E18	Conduit and Tray Layout, Turbine Building - Area 7
E328, Rev. E21	Conduit and Tray Layout, Turbine Building - Area 9
E333, Sheet 1, Rev. E21	Cable Spreading Room Raceway Layout, Plan El. 23'-0"

Safety Evaluations

Safety Evaluation (SE) 2270 for Procedure 2.4.143.1, Revision 0, 12/22/87
SE 2271 for Procedure 2.4.143.2, Revision 0, 12/22/87
Preliminary Evaluation Checklist for Procedure 2.4.143.1, Revision 8

Quality Assurance Documents

Surveillance Report 96-054, Independent Assessment: Shift Manning, 8/8/96
Surveillance Report 96-060, Shutdown with Fire in Fire Area 1.9 Demonstration
Audit Report 00-07, Biennial Fire Protection Program
MCS99-46, Fire Protection 1999 3rd Quarter Self-Assessment

Other Documents

Memorandum to John Pawlak, from Jeff Rogers, dated 4/24/87, re: Required Components for Appendix R
89XM-1-ER-Q, “Updated Fire Hazards Analysis”
T. P. 83-27, Functional Test of the Cable Spreading Room Fire Extinguisher Halon System
T. P. 83-29, Fire Extinguishing Halon System for Cable Spreading Room. Discharge Test and Temperature-Pressure Monitoring
NFPA 20 Standard For the Installation of Centrifugal Fire Pumps, 1967 Edition
Specification M-570, Fire Barrier and Secondary Containment Penetration Seal Systems
Licensee Event Report 50-293/99-011-00, “Postulated Fire in Cable Spreading Room Potentially Affecting Safe Shutdown”

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.