March 14, 2002

Mr. Robert J. Barrett
Vice President, Operations
Entergy Nuclear Operations, Inc.
Indian Point Nuclear Generating Unit 3
295 Broadway, Suite 3
Post Office Box 308
Buchanan, NY 10511-0308

SUBJECT: INDIAN POINT 3 NUCLEAR POWER PLANT - NRC TRIENNIAL FIRE PROTECTION INSPECTION REPORT NO. 50-286/01-012

Dear Mr. Barrett:

On February 21, 2002, the NRC completed a triennial fire protection team inspection at the Indian Point 3 nuclear power plant. The enclosed report documents the inspection findings which were discussed at an exit meeting on February 21, 2002, with 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 inspectors identified one issue of very low safety significance (Green). This issue involved a violation of NRC requirements; however, because of the very low safety significance and because it has been entered into your corrective action program, the NRC is treating this issue 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 20555-0001; and the NRC Resident Inspector at the Indian Point 3 facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room **or** from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/ADAMS.html (the Public Electronic Reading Room).

Sincerely,

/RA/

James C. Linville, Chief
Electrical Branch
Division of Reactor Safety

Docket No.50-286
License No. DPR-64

Enclosure: Inspection Report No. 50-286/01-012

cc w/encl:
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OFFICE	RI:DRS		RI:DRS		RI:DRP		RI:DRS	
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DATE	03/11/02		03/13/02		03/12/02		03/11/02	

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-286
License No.	DPR-64
Report No.	50-286/01-012
Licensee:	Entergy Nuclear Northeast
Facility:	Indian Point 3 Nuclear Power Plant
Location:	295 Broadway, Suite 3 Buchanan, NY 10511-0308
Dates:	February 4 - 21, 2002
Inspectors:	T. Walker, Sr. Reactor Inspector, Division of Reactor Safety (DRS) L. Cheung, Sr. Reactor Inspector, DRS K. Young, Reactor Inspector, DRS L. James, Resident Inspector, Division of Reactor Projects
Approved by:	James C. Linville, Chief Electrical Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000286/2001-012, on 02/04-21/2002, Entergy Nuclear Northeast, Indian Point 3 Nuclear Power Plant. Fire Protection.

The inspection was conducted by a team composed of regional specialists and a resident inspector. The inspection identified one green finding which was a non-cited violation. The significance of most findings 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/reactors/operating/oversight.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 achieve cold shutdown conditions within 72 hours following a fire. The team found that the procedures for shutdown from outside of the control room did not provide sufficient direction to assure that pressurizer pressure could be reduced to allow initiation of the residual heat removal system for decay heat removal in sufficient time to ensure that cold shutdown could be achieved within 72 hours of plant shutdown. A delay in achieving cold shutdown following a fire that required shutdown from outside of the control room was considered a credible impact on safety. This finding was of very low safety significance because the likelihood of a fire that could necessitate a shutdown from outside of the control room and cause a loss of reactor coolant system letdown capability was small. (1RO5.8)

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 Entergy Nuclear Northeast has implemented an adequate fire protection program and that post-fire safe shutdown capabilities have been established and are being properly maintained. The following fire areas were selected for detailed review based on risk insights from the Indian Point Unit 3 (IP3) Individual Plant Evaluation for External Events (IPEEE):

- Cable Spreading Room (Fire Area CTL-3/Fire Zone 11)
- Electrical Tunnels and Electrical Penetration Areas (Fire Area ETN-4/Fire Zones 7A, 60A, 73, 73A, and 74A)
- Auxiliary Feedwater Pump Room (Fire Area AFW-6/Fire Zone 23)
- Primary Auxiliary Building 55' EL - Motor Control Center Area and Charging Pump Cubicles (Fire Area PAB-2/Fire Zones 5, 6, 7, and 17A)

This inspection was a reduced scope inspection in accordance with the March 23, 2001, 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.

- **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, as well as several hot-work permits. These reviews were accomplished to ensure that the licensee was maintaining the fire protection systems, controlling hot-work activities, and controlling combustible materials in accordance with AP-64, "IP3 Site Fire Protection Program," and other fire protection program procedures.

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 licensee submittals and NRC safety evaluation reports (SERs) associated with fire protection features at IP3. The design and qualification testing for raceway fire barriers were also reviewed and a walk-down of installed barriers was performed for the selected fire areas. These reviews were performed to ensure that the passive fire barriers met the licensing and design bases as described in the licensee submittals, NRC SERs, and IP3-ANAL-FP-02143, “Fire Hazards Analysis (FHA) Report.”

The team randomly selected three fire barrier penetration seals for detailed inspection to verify proper installation and qualification. The team reviewed associated design drawings, selected barrier and penetration seal engineering evaluations, and a fire barrier and penetration seal inspection procedure. The team compared the observed in-situ seal configurations to the design drawings and tested or evaluated configurations. 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 as described in the licensee submittals and NRC SERs.

b. Findings

Raceway Protection

At IP3, Hemyc fire wrap is used to protect raceway cable trays and conduit associated with instrumentation cables in the electrical tunnels and electrical penetration areas. The Hemyc fire wrap is provided to meet the separation requirements of 10 CFR 50, Appendix R, Section G, “Fire Protection of Safe Shutdown Capability.” A fire barrier with a one hour fire rating is required to ensure the availability of source range flux instrumentation, as well as channel IV reactor coolant system (RCS) and steam generator instrumentation, for postulated fires in the electrical tunnels and penetration areas. Since the 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 staff 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 IP3, the licensee evaluated the adequacy of the Hemyc fire barrier in evaluation IP3-ANAL-FP-01375, “A Review of 1-hour Fire Barrier Wraps Outside Containment,” Rev. 0. This analysis, which concluded the installations at IP3 provided a one hour fire resistance capability, was based, in part, on the results of the qualification tests discussed in TIA 99-028. These qualification tests are CTP-1026, “‘Hemyc’ Cable Wrap System (Redundant Cable Protection - One Hour),” dated June 1, 1982, and CTP-1077, “One (1) Hour Fire Test on 3" Conduit,” dated March 10, 1986.

The team noted that the Hemyc fire wrap at IP3 was properly maintained and was included in the licensee’s surveillance program for periodic inspections.

The adequacy of the Hemyc barrier at IP3 is unresolved pending further NRC review to determine whether the qualification tests of the Hemyc fire wrap systems are acceptable **(URI 05000286/2001-012-01).**

.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 the type of installed detectors as shown per location drawings. The team also reviewed licensee submittals and NRC SERs associated with the selected fire areas. Additionally, the team reviewed the licensee’s fire protection design basis document (DBD) and the National Fire Protection Association (NFPA) compliance report. 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 as described in the licensee submittals and NRC SERs. The team also reviewed fire detection surveillance procedures and the technical requirements manual (TRM) to determine the adequacy of fire detection component testing and to ensure that the detection systems could function when needed.

b. Findings

No findings of significance were identified.

.4 Fixed Fire Suppression Systems and Equipment

a. Inspection Scope

The team reviewed the adequacy of the carbon dioxide (CO_2) suppression system in the cable spreading room, the pre-action sprinkler systems in the electrical tunnels and penetration areas, and the wet-pipe sprinkler system in the auxiliary feedwater pump room by performing walk-downs of the systems. The team verified suppression system functionality and the adequacy of surveillance procedure testing by reviewing completed surveillance procedures, the TRM, and hydraulic calculations for the sprinkler systems. The team reviewed initial discharge testing, design specifications, minor modifications, calculations and engineering evaluations for the cable spreading room CO_2 suppression system. The team also reviewed and walked down pre-fire plans and CO_2 system operating procedures. These reviews were performed to ensure that the fixed suppression systems in the selected risk significant fire areas met the design and licensing bases as described in the licensee submittals and NRC SERs, and that the systems could perform their intended functions in the event of a fire in the respective areas.

b. Findings

No findings of significance were identified.

.5 Manual Fire Suppression Capability

a. Inspection Scope

The team walked down selected standpipe systems and observed portable extinguishers to determine the material condition of manual fire fighting equipment and verify locations as specified in the pre-fire plans and fire protection program documents. Electric fire pump and diesel fire pump flow and pressure tests were also reviewed by the team to ensure that the pumps were meeting design requirements. The team inspected the fire brigade's protective ensembles, self-contained breathing apparatus (SCBA), and various fire brigade equipment to determine operational readiness for fire fighting.

The team reviewed the fire brigade leader and member training and qualifications to assure that fire fighting personnel were properly trained and qualified. The team verified that selected fire brigade members had participated in a minimum of two fire drills during the calender year 2001 and had current SCBA certification. The team reviewed the licensee's revision to AP-64, "IP3 Site Fire Protection," that incorporated a one-time change to the grace period for yearly retraining of the fire brigade members from the security department. The team also reviewed the fire brigade training manual and fire brigade leader training documents.

The team reviewed pre-fire plans for the selected areas to determine if appropriate information was 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 the IP3 fire suppression effects analysis to determine if a pipe rupture, inadvertent actuation of a suppression system, or manual fire suppression activities in the selected fire areas could inhibit the plant's ability reach a safe shutdown condition. Additionally, the team performed in-plant walk-downs to evaluate the physical configuration of electrical raceways and safe shutdown components in the cable spreading room (CSR), electrical tunnels and penetration areas, and the auxiliary feedwater pump room to determine whether water from manual fire suppression activities in these areas could cause damage that could inhibit the plant's ability to safely shutdown. The team also reviewed the licensee's actions to address the potential for CO_2 migration to ensure that fire suppression and post-fire safe shutdown actions would not be impacted.

b. Findings

No findings of significance were identified.

.6 Safe Shutdown Capability

a. Inspection Scope

The team reviewed the IP3 safe shutdown analysis (SSDA), IP3-ANAL-FP-01503, “Appendix R Sections III.G and III.L Safe Shutdown Analysis Report,” 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, to identify equipment required to isolate flow diversion paths, and to verify appropriate components were identified as required safe shutdown equipment. 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 the deviations had been approved and that conditions required by the deviations were implemented and being maintained.

The team also reviewed selected evaluations and calculations that supported the SSDA to confirm that the loss of primary or support equipment due to a fire would not prevent safe shutdown of the plant. For example, the team reviewed calculations which demonstrated that loss of one switchgear room ventilation fan due to a fire in the electrical tunnels or primary auxiliary building would not adversely impact credited equipment in the switchgear room. The team interviewed licensee technical personnel and confirmed that sufficient margin existed for room cooling with a mix of high temperature outside air and higher temperature air from the turbine building.

b. Findings

No findings of significance were identified.

.7 Safe Shutdown Circuit Analyses

a. Inspection Scope

The team reviewed Section 7, “Safe Shutdown Circuit Identification and Analysis,” of the IP3 SSDA 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 and analyses for selected risk-significant post-fire safe shutdown components, documented in Section 8 of the SSDA, to determine if the cables were properly routed outside the fire areas of concern or protected against the effects of the postulated fires. For example: the pumps, valves, and instrumentation for feeding the steam generators were reviewed for postulated fires in the AFW pump room or the electrical tunnels and penetration areas; and selected motor control centers were reviewed for postulated fires in the primary auxiliary building or the electrical tunnels and penetration areas. The team also walked down portions of cable routing to confirm that the cables required for safe shutdown would not be impacted by the postulated fires.

The team reviewed the Appendix R breaker coordination study, documented in IP3-RPT-ED-00723, "Appendix R Diesel Generator System Evaluation," 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 and preventive maintenance procedures for the Appendix R 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.

.8 Operational Implementation of Safe Shutdown Capability

a. Inspection Scope

The team reviewed post-fire shutdown procedures (off-normal operating procedures and system operating procedures) for the selected areas to determine if appropriate information is provided to plant staff to perform required actions to achieve and maintain safe shutdown. This review included a comparison of the procedures with the SSDA to ensure that the actions assumed in the SSDA were included in the procedures. The team also reviewed training lesson plans and job performance measures (JPMs) for post-fire activities.

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 spreading room and was assumed to cause control room indications and controls to be unreliable, requiring plant shutdown from outside of the control room. In the postulated scenario, equipment failed sequentially, allowing the operators to address the early failures using the off-normal procedure for a fire, ONOP-FP-1, "Plant Fires," and supporting procedures from the control room. Ultimately, the operators determined that the postulated situation warranted shutdown from outside of the control room due to loss of control of multiple safe shutdown systems. The remainder of the actions required to achieve hot shutdown were implemented in accordance with ONOP-FP-1A, "Safe Shutdown From Outside the Control Room."

A licensed senior reactor operator (SRO), a licensed reactor operator (RO), and a non-licensed plant operator (NPO) simulated the actions required to establish hot standby conditions using ONOP-FP-1, ONOP-FP-1A, and other support procedures. The team evaluated whether minimum shift staffing was sufficient to implement ONOP-FP-1, ONOP-FP-1A, and other procedures required to achieve safe shutdown from outside of the control room. The team assessed the accessibility of the alternative shutdown operating stations and the accessibility of required manual action locations. This assessment included consideration of potential CO_2 migration from the cable spreading room suppression system. The team also evaluated fire hazards in the vicinity of equipment requiring operator actions, and along the access and egress paths.

The team also conducted table-top reviews with operators of portions of the post-fire safe shutdown procedures needed to respond to a fire in the electrical tunnel entrance to verify the adequacy of the procedures for establishing hot shutdown conditions from the control room. The inspectors also performed an in-plant walk-down of the procedure for operation of the Appendix R diesel generator with operators to verify procedure adequacy, equipment accessibility, and tool and equipment availability.

b. Findings

The team identified a deficiency in the procedures for transition to cold shutdown from outside of the control room that could have delayed cold shutdown. A delay in achieving cold shutdown following a fire that required shutdown from outside of the control room could have an adverse impact on safety. However, the safety significance of the finding was determined to be very low because there was no impact on fire barriers, fire detection, or fire suppression capability for the designated alternate shutdown areas, and the likelihood of the fire event that could lead to the condition was small. The failure to have adequate procedures to achieve cold shutdown conditions within 72 hours following a fire in an alternate shutdown area was considered a non-cited violation (NCV) of 10 CFR 50, Appendix R, Section III.L.

In 1989, Westinghouse performed a safety evaluation to support an increase in the design basis maximum temperature of the IP3 ultimate heat sink to 95°F. Based on this analysis, it was determined that the plant could achieve cold shutdown conditions within 72 hours as long as the residual heat removal (RHR) system was initiated approximately 29 hours after plant shutdown following an "Appendix R" fire. This limitation was based on the availability of only one RHR pump, one component cooling water (CCW) pump, one charging pump, and one service water (SWS) pump, which would be the minimum equipment guaranteed to be available for a shutdown from outside of the control room following a fire. In 1995, the licensee identified that the more restrictive requirement to initiate RHR within 29 hours after shutdown had not been incorporated into the SSDA (which assumed that RHR would be placed in service 47.5 hours after shutdown) or the associated procedures. At this time, the licensee revised the SSDA to reflect the requirement to initiate RHR within 29 hours and added instructions to ONOP-FP-1B, “Cooldown From Outside the Control Room,” to notify the Technical Support Center (TSC) for assistance if unable to achieve the conditions required to initiate RHR within 29 hours.

Also in 1995, the licensee identified that the requirement to reach RHR initiation conditions within 29 hours after shutdown would be difficult to meet by relying on ambient losses to lower RCS pressure. This difficulty would be due to the loss of RCS letdown capability due to the postulated fire and the need to provide continuous seal injection with a charging pump for reactor coolant pump (RCP)seal integrity until CCW cooling could be restored to the RCP thermal barriers. The licensee’s proposed resolution was to provide a means to reduce pressurizer pressure by either restoration of auxiliary spray by local manual valve operations or development of a repair procedure to allow control of a power-operated relief valve (PORV) for letdown. Restoration of auxiliary spray was selected as the preferred method. This method was added to the SSDA as part of the safe shutdown methodology for transition to cold shutdown. However, no direction for use of auxiliary spray was added to ONOP-FP-1B and no reference to the required local manual valve manipulations was added to the sub-tier fire off-normal procedure ONOP-FP-30, “Control Building Fires - CTL-3.”

The team concluded that there was insufficient procedural direction to ensure that the operators could reduce pressurizer pressure to below 400 psig to allow RHR initiation within 29 hours of plant shutdown in the case of a fire that required shutdown from outside of the control room. Although ONOP-FP-1B provided direction to consult the TSC, there were no written instructions for use of auxiliary spray or specific directions for the required local manual valve manipulations. Nor was there procedural direction for use of the PORVs for letdown or specific directions for the repair that could be required due to fire damage. Additionally, there were several factors that could have delayed or confused the operators in making the determination that RHR conditions could not be achieved within the required time limit. These factors included: 1) cautions related to cooldown time limits at the beginning of ONOP-FP-1B which were less restrictive than the 29 hour limit; 2) no direction in ONOP-FP-1A that would ensure expeditious entry into ONOP-FP-1B to commence cooldown and depressurization and 3) the first reference to the 29 hour limit was in step 13 of ONOP-FP-1B.

The team determined that the failure to provide procedural direction for reduction of pressurizer pressure had a credible impact on safety because it could result in a delay in initiation of RHR for decay heat removal. Under certain conditions, this delay could result in the inability to achieve cold shutdown conditions within 72 hours. Using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process (SDP),” Appendix F, “Determining Potential Risk Significance of Fire Protection and Post-Fire Safe Shutdown Inspection Findings,” the team determined that the procedure deficiency was a degradation of a fire protection feature which could adversely impact the ability to achieve cold shutdown. However, 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 Phase I of the fire protection SDP. The team determined that the failure to provide procedural direction for reduction of pressurizer pressure was of very low safety significance (GREEN), because the likelihood of a fire which could result in the loss of the various pumps and letdown capability, in conjunction with a maximum ultimate heat sink temperature, was small.

10 CFR 50, Appendix R, Section III.L, “Alternative and Dedicated Shutdown Capability,” requires, in part, that procedures be in effect to implement the capability to achieve cold shutdown conditions within 72 hours for fire areas required to have alternate or dedicated shutdown capability. Contrary to this requirement, procedures were not in effect to assure that cold shutdown conditions could be achieved within 72 hours under certain conditions for a fire in the control building, an area required to have alternate or dedicated shutdown capability. This violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy **(NCV 05000286/2001-012-02)**. The inconsistency between the SSDA, which credits auxiliary spray for pressure reduction, and the operating procedures is in the licensee’s corrective action program as DER-02-00540.

.9 Post-Fire Safe Shutdown Emergency Lighting and Communications

a. Inspection Scope

The team observed the placement and aim of emergency battery light (EBL) units 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, and observed equipment operation during procedure walk-downs 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, surveillance procedures and vendor information to determine if adequate surveillance testing was being accomplished to ensure operation of the emergency lights. Additionally, the team reviewed a design change package (DCP) 99-096 EML, “Replacement of Appendix R Emergency Battery Lights,” to determine if the licensee had addressed battery issues for high temperature environments.

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 self-assessment reports and quality assurance audit reports for fire protection activities conducted during the past two years. Selected deviation/event reports (DERs) for fire protection and post-fire safe shutdown equipment were also reviewed. This review included the DERs initiated to address issues identified during this inspection. The team also reviewed selected outstanding and completed fire protection equipment work requests. These reviews were conducted to determine if Entergy Nuclear Northeast was identifying fire protection 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. Barrett and other members of the Entergy Nuclear Northeast staff at an exit meeting on February 21, 2002.

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.

ATTACHMENT 1
SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Entergy Nuclear Northeast

R. Barrett, Vice President - Operations
J. Bencivenga, HVAC Engineer
R. Christman, Assistant Operations Manager
J. Comiotes, Director Nuclear Safety Assurance
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R. Schimpf, Sr. I & C Design Engineer
S. VanBuren, Fire Protection Supervisor
G. Vranjesevic, Sr. Electrical Design Engineer
S. Wilkie, Fire Protection Engineer

Nuclear Regulatory Commission

P. Drysdale, Senior Resident Inspector
J. Linville, Chief, Electrical Branch

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000286/2001-012-02	NCV	Inadequate Procedure for Transition to Cold Shutdown During Shutdown From Outside the Control Room (Section 1R05.8)

Opened

05000286/2001-012-01	URI	Adequacy of Hemyc Cable Wrap Fire Barrier Test and Evaluation (Section 1R05.2)

Closed

None

Discussed

None

LIST OF ACRONYMS USED

AFW	Auxiliary Feedwater
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CO_2	Carbon Dioxide
CSR	Cable Spreading Room
DBD	Design Basis Document
DCP	Design Change Package
DER	Deviation/Event Report
EBL	Emergency Battery Light
FHA	Fire Hazards Analysis
IMC	Inspection Manual Chapter
IPEEE	Individual Plant Evaluation for External Events
IP	Inspection Procedure
IP3	Indian Point 3
IR	Inspection Report
JPM	Job Performance Measure
NCV	Non-Cited Violation
NFPA	National Fire Protection Association
NPO	Non-Licensed Plant Operator
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
ONOP	Off-Normal Operating Procedure
P&ID	Piping and Instrumentation Drawing
PAB	Primary Auxiliary Building
PORV	Power-Operated Relief Valve
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RO	Reactor Operator
SCBA	Self Contained Breathing Apparatus
SDP	Significance Determination Process
SER	Safety Evaluation Report
SRO	Senior Reactor Operator
SSDA	Safe Shutdown Analysis
SWS	Service Water System
TIA	Task Interface Agreement
TRM	Technical Requirements Manual
TSC	Technical Support Center
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item

LIST OF DOCUMENTS REVIEWED

Piping and Instrumentation and Design Drawings (P&ID)

9321-F-20173, "Flow Diagram - Main Steam," Rev. 62
9321-F-20183, "Flow Diagram - Condensate & Boiler Feed Pump Suction," Rev. 55
9321-F-20193, "Flow Diagram - Boiler Feedwater," Rev. 56
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Unannounced Fire Drill at the 32 Rod Drive Motor Generator Set on 5/21/00 at 2311 hours
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Unannounced Fire Drill at the 31 Auxiliary Boiler Feedpump on 6/30/99 at 0807 hours
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DER-00-00136	DER-00-00410	DER-00-00503	DER-00-01485
DER-00-01939	DER-00-02289	DER-00-02293	DER-00-02309

DER 00-02321	DER-00-02322	DER-00-02351	DER-00-02435
DER-00-02656	DER-00-02869	DER-00-02905	DER-00-03139
DER-00-03305	DER-01-00991	DER-01-01071	DER-01-01135
DER-01-01173	DER-01-01200	DER-01-01220	DER-01-01222
DER-01-01461	DER-01-01484	DER-01-01765	DER-01-02863
DER-01-03062	DER-01-03075	DER-01-03079	DER-01-03100
DER-01-03101	DER-01-03214	DER-01-03294	DER-01-03333
DER-01-03427	DER-01-03520	DER-01-03521	DER-01-03545
DER-01-03546	DER-01-03740	DER-01-03864	DER-01-04099
DER-01-04171	DER-01-04245	DER-01-04434	DER-01-04454
DER-02-00001	DER-02-00060	DER-02-00420*	DER 02-00421*
DER-02-00433*	DER-02-00540*	DER-02-00546*	DER 02-00549*
DER 02-00556*	DER 02-00558*		

* Denotes DERs initiated during the inspection

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WR No. 96-03822-09	WR No. 00-02252-00	WR No. 01-00026-03
WR No. 97-04633-00	WR No. 00-02815-00	WR No. 01-00054-00
WR No. 99-05022-30	WR No. 00-02816-00	WR No. 01-03406-00
WR No. 99-05022-31	WR No. 00-03144-00	WR No. 01-04051-00
WR No. 99-05293-02	WR No. 00-04800-00	WR No. 01-04849-00
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