August 20, 2004

Mr. Fred Dacimo
Site Vice President
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
295 Broadway, Suite 1
P.O. Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 - NRC TRIENNIAL FIRE PROTECTION INSPECTION REPORT 05000247/2004005

Dear Mr. Dacimo:

On March 19, 2004, the US Nuclear Regulatory Commission (NRC) completed a triennial fire protection inspection at your Indian Point Unit 2 facility. The enclosed report documents the inspection findings, which were discussed on July 20, 2004, 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 rules and regulations, and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection no findings of significance were identified. However, licensee-identified violations which were determined to be of very low safety significance are listed in Section 40A7 of this report. 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 1; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Indian Point Unit 2.

In accordance with 10 CFR 2.390 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 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).

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

Sincerely,

/RA/

John F. Rogge, Chief
Electrical Branch
Division of Reactor Safety

Docket Nos. 50-247
License Nos. DPR-26

Enclosure: NRC Inspection Report 05000247/2004005
w/Attachment: Supplemental Information

cc w/encl:
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U.S. NUCLEAR REGULATORY COMMISSION

REGION 1

Docket No:	50-247
License No:	DPR-26
Report No:	05000247/2004005
Licensee:	Entergy Nuclear Northeast
Facility:	Indian Point Nuclear Generating Unit No. 2
Location:	Buchanan, NY
Inspection Dates:	March 1 - 19, 2004 (In office inspection follow-up through April 16, 2004)
Inspectors:	Keith Young, Senior Reactor Inspector, DRS (Team Leader) Larry Scholl, Senior Reactor Inspector, DRS Timothy O'Hara, Reactor Inspector, DRS Jonathan Lilliendahl, Reactor Inspector, DRS (Observer)
Approved by:	John F. Rogge, Chief Electrical Branch Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000247/2004005; 03/01/2004 - 03/19/2004; Indian Point Nuclear Generating Unit No. 2; Triennial Fire Protection Inspection. In-office inspection follow-up activities occurred through April 16, 2004.

The report covered a two week team inspection by specialist inspectors. The team identified one unresolved item. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. <u>NRC-Identified Findings</u>

No findings of significance were identified.

B. <u>Licensee-Identified Violations</u>

Violations of very low safety significance, which were identified by Entergy Nuclear Northeast have been reviewed by the team. Corrective actions taken or planned by Entergy Nuclear Northeast have been entered into their corrective action program. These violations and corrective actions are listed in Sections 4OA3 and 4OA7 of this report.

Report Details

Background

This report presents the results of a triennial fire protection 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 at the Indian Point Nuclear Generating Unit No. 2. The following fire zones (FZ) and a fire area (FA) were selected for detailed review based on risk insights from the Indian Point Nuclear Generating Station, Unit 2, Individual Plant Examination (IPE) and Individual Plant Examination of External Events (IPEEE):

- Fire Zone 10 - Diesel Generator Building
- Fire Zone 14 - 480 V Switchgear Room
- Fire Zone 15 - Central Control Room
- Fire Zone 32A - Electrical Tunnel
- Fire Area J - Turbine Building

This inspection was a reduced scope inspection in accordance with the March 6, 2003, 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 NRC Generic Letter (GL) 81-12.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems

1R05 Fire Protection

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. These reviews were accomplished to ensure that the Entergy Nuclear Northeast was maintaining the fire protection systems, controlling hot-work activities, and controlling combustible materials in accordance with the Updated Final Safety Analysis Report (UFSAR)/Fire Protection Program Plan (FPPP), administrative procedures and other fire protection program procedures.

b. Findings

No findings of significance were identified.

2. Fire Area Boundaries and 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 Indian Point Unit 2. Additionally, the team reviewed the design and qualification testing of selected barriers and reviewed surveillance procedures for structural fire barriers, penetration seals and structural steel. These reviews were performed to ensure that the passive fire barriers were properly maintained and met the licensing and design bases as described in the licensee submittals, NRC SERs, the Indian Point Unit 2, UFSAR and the FPPP.

b. Findings

No findings of significance were identified.

3. Post-Fire Safe Shutdown Lighting and Communications

a. Inspection Scope

The team observed the placement and aim of eight-hour emergency lights throughout the selected fire areas to evaluate their adequacy for illuminating access and egress pathways and any equipment requiring local operation for post-fire safe shutdown. The team also reviewed preventive maintenance procedures and various documents, including the vendor manual and surveillance tests, to determine if adequate surveillance testing and periodic battery replacements were in place to ensure reliable operation of the emergency lights.

The team reviewed radio repeater location and power sources to ensure fire brigade and operator communications could be maintained for fire fighting and post-fire safe shutdown conditions.

b. Findings

No findings of significance were identified.

4. Fire Detection Systems and Equipment

a. Inspection Scope

The team reviewed the adequacy of the fire detection systems in the selected plant fire areas. This included a walkdown of the systems and review of the type of installed detectors as shown per location drawings. The team also reviewed licensee submittals, engineering evaluations and the NRC SERs associated with the selected fire areas. These reviews were performed to ensure that the fire detection systems for the selected fire areas were installed in accordance with the design and licensing bases of the plant. Additionally, the team reviewed fire detection surveillance procedures to determine the adequacy of the fire detection component testing and to ensure that the detection system would function as required.

b. Findings

No findings of significance were identified.

5. Fixed Fire Suppression Systems

a. Inspection Scope

Sprinkler Systems and Equipment

The team reviewed the adequacy of the pre-action sprinkler system for FZ 32A, Electrical Tunnel, and the wet pipe sprinkler system in FZ 10, Diesel Generator Room, by performing a walkdown and observing the material condition of the system. Additionally, the team reviewed the design and installation specifications, NFPA 13, "Standard for the Installation of Sprinkler Systems," installation drawings, the adequacy of surveillance procedures and hydraulic calculations. These reviews were performed to ensure that the sprinkler system met the design and licensing bases as described in the licensee submittals, NRC SERs and the UFSAR/FPPP, and that the systems could perform their intended function in the event of a fire in their respective areas.

b. Findings

No findings of significance were identified.

6. Manual Fire Suppression Capability

a. Inspection Scope

The team walked down selected standpipe systems and observed portable extinguishers to determine the material condition of the manual fire fighting equipment and verify locations as specified in the pre-fire plans and fire protection program documents. The team reviewed electric and diesel fire pump flow and pressure tests to ensure that the pumps were meeting their design requirements. The team also reviewed the fire main loop flow tests to ensure that the flow distribution paths were able to meet the 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 pre-fire plans and smoke removal strategies for the selected fire 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 performed in-plant walkdowns to evaluate the physical configuration of electrical raceway and safe shutdown components in the selected fire areas to determine whether water from an inadvertent fire suppression system pipe rupture or from manual fire suppression activities in the selected areas could cause damage that could inhibit the ability to safely shut down the plant.

The team reviewed fire brigade initial training and continuing training course materials to verify appropriate training was being conducted for the station firefighting personnel. Additionally, the team reviewed selected fire drills and critiques to ensure that drills were being conducted in risk significant areas.

The team reviewed the qualifications of several fire brigade leaders and members to ensure that they had met and maintained the requirements to be qualified fire brigade members.

b. Findings

No findings of significance were identified.

7. Post-Fire Safe Shutdown Capability

a. Inspection Scope

The team reviewed the fire response procedures, alarm response procedures, operating procedures for the selected fire areas, and the FPPP to evaluate the methods and equipment used to achieve hot shutdown following a fire. The team also reviewed piping and instrumentation drawings (P&ID) for post-fire safe shutdown systems to identify required components for establishing flow paths, to identify equipment required to isolate flow diversion paths, and to verify appropriate components were properly evaluated and included in the safe shutdown equipment list. The team also reviewed selected alternate safe shutdown components and their control circuits to ensure that proper isolation was provided for alternate safe shutdown capability, in the event of a fire affecting the control room, cable tunnel or the 480 V switchgear room. The team performed field walkdowns to evaluate the protection of the equipment from the effects of fires.

A post-fire safe shutdown procedure for the selected areas was also reviewed to determine if appropriate information was provided to plant operators to identify protected equipment and instrumentation and if recovery actions specified in the post-fire safe shutdown procedure considered manpower needs for performing required actions. The team also reviewed training lesson plans for the alternative shutdown procedures, discussed training with licensed operators, reviewed selected alternate safe shutdown equipment tests and calibrations, reviewed the adequacy of shift manning, and evaluated the accessibility of the alternative shutdown operating stations and required manual action locations.

The specific alternate safe shutdown procedure reviewed was 2-AOP-SSD-1, “Control Room Inaccessibility Safe Shutdown Control,” Revision 1. A procedure walkdown was performed for this procedure. The walkdown was performed by an operations crew and focused primarily on the portion of the procedure associated with achieving stable hot shutdown conditions. Plant operators were accompanied by NRC team members during the walkdown and the approximate time for critical steps, such as establishing makeup flow to the reactor vessel, aligning the alternate safe shutdown system (ASSS) to supply electrical power to safe shutdown loads and establishing component cooling water (CCW) for reactor coolant pump (RCP) seal cooling were noted and evaluated to assess the ability of the operators to maintain plant parameters within procedural limits.

b. Findings

Introduction. The team identified and unresolved item concerning the ability of the alternate safe shutdown procedure 2-AOP-SSD-1, “Control Room Inaccessibility Safe Shutdown Control,” Revision 1, to ensure a safe shutdown of the plant for a severe fire in the 480 volt switchgear room. Specifically, the procedure may not have precluded an extended loss of reactor coolant pump seal injection flow which could have led to RCP leakage above the Westinghouse Owners Group (WOG) guidance of 21 gallons per minute after 13 minutes. This issue will remain unresolved pending further NRC review of information regarding RCP seal performance following a loss of all seal cooling event.

Description. Following review and a walkdown of Revision 1 of procedure AOP-SSD-1, the team noted the following issues that reduced its efficiency. The team’s observations were as follows:

a. The procedure did not prioritize actions to ensure prompt restoration of electrical power and RCP seal cooling via seal injection and/or CCW flow to the seal coolers.

b. After GT-1 is started, the procedure verifies if CCW and/or a charging pump is operating. If they are not running, instead of starting the pumps on ASSS power (GT-1), the procedure had operators preparing to start a safety injection (SI) pump. The SI pump requires casualty cable installation and would not be immediately available.

c. The step to start a charging pump on ASSS power came after initiating the SI pump actions.

d. Based on their training, the operators assumed RCP seal damage may occur after 15 minutes without RCP seal injection and, therefore, the procedure focus on SI pump recovery seemed to make sense to them.

e. On March 19, 2004, Entergy Nuclear Northeast discovered that GT-1 would not have started locally because of a failed op-trend computer in the control room.

The team expressed these and other concerns to plant personnel. Entergy Nuclear Northeast initiated CR-2004-1445 to complete a recovery plan for the identified procedural concerns. The recovery plan included implementing compensatory measures, revising the FPPP, revising procedure AOP-SSD-1 and performing a calculation to ensure the ASSS procedure steps could be performed to meet the performance goals for 10 CFR 50, Appendix R. Additionally, Entergy Northeast performed timeline validations of the revised procedure.

Based on the observed walkdown of Revision 1 of procedure AOP-SSD-1, the team estimated that more than 60 minutes could elapse without charging or CCW before the ASSS would be aligned to provide electrical power to these systems. Current WOG guidance suggest that this prolonged loss of electrical power, that supplies RCP seal cooling, could cause increased seal leakage after 13 minutes. After some time, RCP seal leakage could become excessive. If RCP seal cooling is not established within 13 minutes following the loss of electrical power, current WOG guidance suggests that it should not be established prior to performing a cooldown to protect the seals. Additionally, the FPPP references NUREG/CR-2934, Review and Evaluation of the Indian Point Probabilistic Safety Study, which states that seal failure is assumed to occur following failure of the redundant means of providing cooling (i.e., charging and the component cooling system) and is predicted to lead to significant seal leakage at 30 minutes. The Safety Evaluation of Topical Report WCAP-15603, Revision 1, "WOG 2000 Reactor Coolant Pump Seal Leakage Model For Westinghouse PWRs," indicates that leakage of RCP seal could increase to more than 21 gpm after 13 minutes.

The identified issue could adversely impacted the efficiency of Entergy Nuclear Northeast's safe shutdown strategy and could have prevented achieving and maintaining hot shutdown in the post-fire environment. Because this issue could affected the reactor safety mitigating system cornerstone objective, the issue could be greater than minor. The safety significance of this finding was not evaluated due to further NRC review of RCP seal leakage issues.

It appeared that Entergy Nuclear Northeast did not provide an alternate safe shutdown procedure AOP-SSD-1, Revision 1, that would efficiently protect RCP seals from potentially leaking more than the 21 gpm after 13 minutes per NRC safety evaluation guidance and the guidance provided in NUREG/CR-2934, that excessive RCP seal leakage could occur after 30 minutes. Revision 4 of the alternate safe shutdown procedure appeared to incorporate several improvements that increased the efficiency of accomplishing the necessary safe shutdown tasks. The team concluded that the identified issue concerning potential effects on alternate safe shutdown for RCP seals is an unresolved item (URI) pending further NRC review of information regarding RCP seal performance following a loss of all seal cooling event. This issue is identified as **URI 05000247/2004005-01, RCP Seal Cooling for Alternate Safe Shutdown Procedure Strategies**.

8. <u>Safe Shutdown Circuits</u>

a. <u>Inspection Scope</u>

The team reviewed power and control cable routing for a sample of components required for post-fire safe shutdown to determine if the cables were properly evaluated as part of the safe shutdown analyses in the UFSAR/FPPP. The team performed a walkdown of accessible selected alternate safe shutdown system cables to ensure cables were not routed in the inspected fire areas. Additionally, the team reviewed cable routing for selected safe shutdown components. This was accomplished by reviewing cable routing drawings and performing infield verification of accessible cables to ensure they were not routed in the inspected fire areas.

The team reviewed electrical fuse and circuit breaker coordination studies to ensure equipment needed to conduct post-fire safe shutdown activities would not be impacted due to a lack of coordination. The team confirmed that coordination studies had addressed multiple faults due to fire. The team also 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.

Additionally, the team reviewed calculations and procedures required to ensure minimum operation voltages were being supplied to alternate safe shutdown equipment.

Due to the issuance of Change Notice 00-020 to 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 the open corrective maintenance work orders for fire protection and safe shutdown equipment, selected condition reports (CRs) for fire protection and safe shutdown issues and the fourth quarter (2002) Unit 2 fire protection systems health report to evaluate the prioritization for resolving fire protection related deficiencies and the effectiveness of corrective actions. The team also reviewed recent Quality Assurance (QA) audits and self-assessments of the fire protection program to determine if the licensee was identifying program deficiencies and implementing appropriate corrective actions.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

1. (Closed) LER 05000247/2003001-00, Plant is an Unanalyzed Condition due to Cable Routing Non-Compliance with Appendix R Separation Criteria

On February 6, 2003, Entergy Nuclear Northeast identified that the routing of charging pump control cables did not meet the cable separation criteria specified in 10 CFR 50, Appendix R, section III.G. On February 12, 2003, Entergy Nuclear Northeast further identified that the routing of the power supply cables for the six service water pumps did not meet the separation criteria of 10 CFR 50, Appendix R, section III.G. These issues were identified during validation and re-baselining of the10 CFR 50, Appendix R, post-fire safe shutdown analysis. It was determined that the identified cables would be vulnerable to fire damage in the fire areas that the cables are routed. Charging pump cables would be vulnerable to a fire in the primary auxiliary building (fire area F) and service water pump cables would be vulnerable to a fire in the turbine building (fire area J). Entergy Nuclear Northeast determined the apparent cause of these issues was lack of rigor applied to the engineering analysis that developed the post fire safe shutdown analysis. Specifically, the engineering analysis lacked sufficient detail and/or support documentation to justify the original design configuration. Corrective actions included implementation of compensatory actions upon discovery, completion of re-baselining of the Appendix R analysis and implementation of plant modifications (planned modifications) to resolve non-compliance issues with Appendix R. These findings are more than minor because they had a credible impact on safety, in that if a fire had occurred in the identified areas, systems used to shut down the plant could have been challenged. These findings affected the Mitigating Systems Cornerstone and were considered to have a very low safety significance (Green). The Region I, Senior Risk Analyst reviewed and agreed with Entergy Nuclear Northeast's determination of risk associated with the charging pump issue. This risk was determined to be of very low significance. The team used Appendix F of the fire protection significance determination process (SDP) to determine the risk associated with the service water issue and the risk was determined to be of very low safety significance. In both cases, the ASSS system was available to shutdown the plant in the event that a fire challenged the systems. Additionally, developing a credible fire in the identified areas to damage all cables is of low probability. These licensee identified findings involved violations of 10 CFR 50, Appendix R, section III. G., cable separation criteria. This LER is closed.

4OA5 Other

(Closed) URI 05000247/2003003-01: Lack of Cable Separation in Fire Areas F and J, Postulated Fire Compromising Associated Circuits

This unresolved item (URI) was opened in NRC Inspection Report 05000247/200303. The team reviewed this URI and the associated LER 05000247/2003001-00. This URI is closed based on the disposition of LER 05000247/2003001-00 which is described in section 4OA3 of this report.

4OA6 Meetings, including Exit

The team presented their preliminary inspection results to Mr. C. Schwarz, GM Plant Operations, and other members of the Indian Point Nuclear Generating Unit No. 2 staff at a debrief meeting on March 18, 2004, and to Mr. F. Dacimo, Site Vice President, at the exit meeting on July 20, 2004. The findings were acknowledged by Entergy Nuclear Northeast. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

4OA7 Licensee-Identified Violations

The following violations of very low safety significance (Green), which were identified by the licensee, were reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the Entergy Nuclear Northeast's corrective action program.

- License Condition 2K requires Entergy Nuclear Northeast to implement and maintain all provisions of the NRC approved fire protection program as described in the UFSAR for the facility and as approved in the SERs dated November 30, 1977, February 3, 1978, January 31, 1979, October 31, 1980, August 22, 1983, March 30, 1984, October 16, 1984, September 16, 1985, November 13, 1985, March 4, 1987, January 12, 1989, and March 26, 1996. By SER dated March 30, 1984, the NRC completed it's review of the licensee's submittal (and exemptions) and concluded that the licensee was in compliance with 10 CFR 50.48, Appendix R, Sections III.G, and Section III.L. 10 CFR 50, Appendix R Section III. G. 1.(a) & (b) requires fire protection features be provided for systems, structures, and components important to safe shutdown. These features shall be capable of limiting fire damage so that: (a) One train of systems necessary to achieve and maintain hot shutdown conditions from either the control room or emergency control station(s) is free from fire damage; and (b) Systems necessary to achieve and maintain cold shutdown from either the control room or emergency control station(s) can be repaired within 72 hours. Contrary to this, the Entergy Nuclear Northeast identified, via CR-IP2-2003-06834 and CR-IP2-2003-02428, that the pneumatic tubing on the steam generator atmospheric steam dump valves contained soldered copper fittings which could be vulnerable to failure if exposed to the high temperature environment of a postulated fire. Such a failure could result in a common-mode failure of all pneumatic lines supplying all four steam generator atmospheric steam dump valves. This failure could disable the atmospheric steam dump valve positioners from both instrument air and nitrogen, thus disabling all steam generator dump valves which are credited for use during alternate safe shutdown system operation for post fire safe-shutdown events. This licensee identified finding is of very low safety significance because the likelihood of occurrence of a fire that could damage all tubing in this area is small, there were no significant combustibles in the area and no loss of post-fire safe-shutdown capability occurred. Additionally, compensatory measures were established pending resolution of this issue. This issue screened to green in phase I of the reactor SDP, MC0609, Appendix A, because it did not involve the total loss of any safety function that would have contributed to identified external event initiated core damage accident sequences.

- License Condition 2K requires Entergy Nuclear Northeast to implement and maintain all provisions of the NRC approved fire protection program as described in the UFSAR for the facility and as approved in the SERs dated November 30, 1977, February 3, 1978, January 31, 1979, October 31, 1980, August 22, 1983, March 30, 1984, October 16, 1984, September 16, 1985, November 13, 1985, March 4, 1987, January 12, 1989, and March 26, 1996. By SER dated March 30, 1984, the NRC completed it's review of the licensee's submittal (and exemptions) and concluded that the licensee was in compliance with 10 CFR 50.48, Appendix R, Sections III.G, and Section III.L. 10 CFR 50, Appendix R, Section III. G. 2., requires, except as provided for in paragraph G.3 of this section, where cables or equipment, including associated non-safety circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided. Contrary to this, Entergy Nuclear Northeast identified, via CR-IP2-2003-01508, that control cables for the containment nitrogen header vent valve HCV-943 were vulnerable to fire-induced faults for fires in Fire Areas A and H. The potential fire-induced faults could cause spurious opening of the valve HCV-943, causing the nitrogen supply to all pneumatically-operated ASSS instruments to be lost. This failure could cause steam generator level, pressurizer level and pressurizer pressure indications to be lost which could disable the ASSS indications. This licensee identified finding is very low safety significance because the likelihood of occurrence of a fire that could damage control cables for these valves is small, procedural steps in the alternate safe shutdown procedure exist to re-enable the ASSS instruments and no loss of safety function occurred. Additionally, compensatory measures were established pending resolution of this issue. This issue screened to green in phase I of the reactor SDP, MC0609, Appendix A, because it did not involve the total loss of any safety function that would have contributed to identified external event initiated core damage accident sequences.

- License Condition 2K requires Entergy Nuclear Northeast to implement and maintain all provisions of the NRC approved fire protection program as described in the UFSAR for the facility and as approved in the SERs dated November 30, 1977, February 3, 1978, January 31, 1979, October 31, 1980, August 22, 1983, March 30, 1984, October 16, 1984, September 16, 1985, November 13, 1985, March 4, 1987, January 12, 1989, and March 26, 1996. By SER dated March 30, 1984, the NRC completed it's review of the licensee's submittal (and exemptions) and concluded that the licensee was in compliance with 10 CFR 50.48, Appendix R, Sections III.G, and Section III.L. 10 CFR 50, Appendix R, Section III. G. 2., requires, except as provided for in paragraph G.3 of this section, where cables or equipment, including associated non-safety circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided. Contrary to this, the Entergy Nuclear Northeast identified, via CR-IP2-2003-06341, that the control cables for charging pump suction path valves LCV-112B and LCV-112C were susceptible to spurious operation (closure). Closure of these valves could result in isolation of both normal and emergency suction paths to the charging pumps. A fire in certain areas of the plant could result in damage to any running charging pumps at the start of the event. Potential pump damage could render the affected charging pumps inoperable from either normal or ASSS power sources, thus potentially disabling all charging capability which is credited in support of

Appendix R performance criteria. This licensee identified finding is of very low safety significance because the likelihood of occurrence of a fire that could damage control cables for these valves is small and no actual loss of safety function occurred. Additionally, compensatory measures were established pending resolution of this issue. This issue screened to green in phase I of the reactor SDP, MC0609, Appendix A, because it did not involve the total loss of any safety function that would have contributed to identified external event initiated core damage accident sequences.

ATTACHMENT: SUPPLEMENTAL INFORMATION

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

F. Dacimo	Site Vice President
C. Schwarz	Admin.-GM Plant Operations
V. Andreozzi	Electrical Design Manager
F. Bloise	DBI Project
P. Conroy	Licensing Manager
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G. Dahl	Licensing
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M. Garofelo	QA Supervisor
P. Gropp	DBI Project Manager
T. Jones	Licensing
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L. Lubrano	Electrical Maintenance
R. Milici	Design Engineering
T. Orlando	PCE Manager
S. Petrosi	Design System Manager
J. Raffaele	Electrical Design Engineering Supervisor
J. Reynolds	CA&A
H. Robinson	Electrical Design Engineer
S. VanBuren	Fire Chief
J. Ventosa	Site Operations Manager
T. Williams	IP2 Ops Manager
S. Wilkie	Fire Protection Engineer

NRC Personnel

W. Lanning, Director, Division of Reactor Safety
J. Rogge, Chief, Electrical Engineering Branch
P. Drysdale, Senior Resident Inspector, Indian Point Unit 2
M. Cox, Resident Inspector, Indian Point Unit 2

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000247/2004005-01	URI	RCP Seal Cooling for Alternate Safe Shutdown Procedure Strategies.

Opened and Closed

NONE

Closed

05000247/2003003-01	URI	Lack of Cable Separation in Fire Areas F and J, Postulated Fire Compromising Associated Circuits
05000247/2003001-00,	LER	Plant in an Unanalyzed Condition due to Cable Routing Non-Compliance with Appendix R Separation Criteria

Discussed

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ENN-LI-102,	Corrective Action Process, Rev. 0
ENN-LI-104,	Self-Assessment and Benchmark Process, Rev. 4
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PC-R37,	Alternate Safe Shutdown Instrument Calibration Check, Rev. 9
SAO-702,	Control of Ignition Sources, Rev. 9
SAO-703,	Fire Protection Impairment Criteria and Surveillance, Rev. 12
SAO-704,	Removal/Reinstallation Fire Rated Assemblies, Rev. 5
SAO-705,	Fire Watch/Fire Watch Tour, Rev. 10
SAO-711,	Quality Assurance Requirements For Fire Protection Systems, Rev. 0
TNG-AD-16,	Fire Protection Training Program, Rev. 6
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2AOP-SSD-1,	Control Room Inaccessibility Safe Shutdown Control, Rev. 1
2AOP-SSD-1,	Control Room Inaccessibility Safe Shutdown Control, Rev. 2
2AOP-SSD-1,	Control Room Inaccessibility Safe Shutdown Control, Rev. 4
2-OSAL 15.13,	Abnormal Breaker Operation, Rev. 1

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IP2 Maintenance Procedure Number BRK-B-004-N,	ITE Type KC, Model G-Air Circuit Breaker - Non-Class "A", Rev. 10
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PT-A17B, Fire Hose Stations, Rev. 6
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PT-A28, Emergency Light Battery Integrity, Rev. 8
PT-EM15, Diesel Generator Building Water Sprinkler System, Rev. 4
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W.O. IP2-01-21490, Perform ICPM-1346 ACS/CCW Return Charging Pump Fluid Drives FI-637, Completed July 18, 2003
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A141537-36,	IP1 Nuclear Service Building Installation Of PWR & Cont. Cond. Elev. 53'-0" & 62'-0" - Plan, Rev. 36
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A206854-21,	IP2 Conduit & Tray Connection Schematic Primary Auxiliary Building EL. 80'-0", Sheet 1, Rev. 21
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A208168-53,	Flow Diagram Chemical & Volume Control System, Rev. 53
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A209762-65,	Flow Diag. Service Wtr Sys Nuclear Stm Supply Plant Sh. 2 of 2, Rev. 65
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A231592-18,	6900 Volt AC One Line Diagram, Rev. 18
A235296-60,	Flow Diagram Safety Injection System, Rev. 60
A244016-19,	One Line 440 VAC Swgr. Unit-Subst’s 11RW1, 12RW3, 12FD3, MCC’s 10M, 10N, 10Z & 10X, Rev. 19
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400402-00,	Fire Area/Zone Arrangement Plans at El. 33'-0", 33'-9", 46'-0", 53'-0", 58'-0", Rev. 0
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9321-2018-136,	Flow Diagram Condensate & Boiler Feed Pump Suction, Rev. 136
9321-F-2019-110,	Flow Diagram Boiler Feedwater, Rev. 110
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W.O. IP2-03-24046, 15' THB
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LIST OF ACRONYMS USED

ASSS	Alternate Safe Shutdown System
CA	Corrective Action
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
DRS	Division of Reactor Safety
FA	Fire Area
FPPP	Fire Protection Program Plan
FZ	Fire Zone
GL	Generic Letter
gpm	Gallons Per Minute
ICA	Implement Corrective Action
IP	Inspection Procedure
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
IR	Inspection Report
LER	Licensee Event Report
MOV	Motor Operated Valve
NCV	Non-Cited Violation
NFPA	National Fire Protection Association
NRC	Nuclear Regulatory Commission
PAR	Publicly Available Record
P&ID	Piping and Instrumentation Drawing
PORV	Power Operated Relief Valve
PWR	Pressurized Water Reactor
QA	Quality Assurance
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
SCBA	Self-Contained Breathing Apparatus
SDP	Significance Determination Process
SER	Safety Evaluation Report
SI	Safety Injection
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
V	Volt
WOG	Westinghouse Owners Group