U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos.:	50-373, 50-374
License Nos.:	NPF-11, NPF-18
Report No.:	50-373/97004; 50-374/97004
Licensee:	Commonwealth Edison Company
Facility:	LaSalle County Station, Units 1 and 2
Location:	2601 N. 21st Road Marseilles, IL 61341
Dates:	February 12, 1997 - May 16, 1997
inspectors:	E. Duncan, Reactor Engineer D. Chyu, Reactor Engineer
Approved by:	M. Ring, Chief, Lead Engineers Branch Division of Reactor Safety

EXECUTIVE SUMMARY

LaSalle County Station, Units 1 and 2
NRC Inspection Report 50-373/97004(DRS); 50-374/97004(DRS)

Engineering

- The inspector reviewed Unresolved Item 50-373/374/96011-12 regarding emergency core cooling system (ECCS) room cooler (VY) flow concerns. The inspector concluded that pre-operational tests regarding ECCS room cooler flow measurement failed to incorporate the effects of residual heat removal service water (RHRSW) system operation on testing results as required by 10 CFR 50, Appendix B, Criterion XI.

- The inspector reviewed Unresolved Item 50-373/374/96011-16 regarding the effect of lake level on surveillance testing acceptance criteria. The inspector concluded that surveillance test LOS-RH-Q1, "RHR (LPCI) and RHR Service Water Pump and Valve Inservice Test for Operational Conditions 1, 2, 3, 4 and 5," failed to include appropriate quantitative acceptance criteria for verifying adequate RHRSW flow as required by 10 CFR 50, Appendix B, Criterion XI.

- The inspector reviewed Unresolved Item 50-373/374/96011-19 regarding a residual heat removal (RHR) heat exchanger waterhammer concern during RHRSW system startup. The inspector determined that significant concerns regarding waterhammer on division 2 of the RHRSW system had been previously identified by the licensee due to a high vertical piping loop unique to division 2 and the licensee planned to install a keep-fill modification for division 2. In addition, the licensee evaluated the effects of potential tube voiding on division 1 and concluded that the stresses imposed on the system due to a potential waterhammer were acceptable. The inspector obtained this evaluation and forwarded it to the Office of Nuclear Reactor Regulation (NRR) for review.

- The inspectors reviewed Unresolved Item 50-373/374/96011-20 regarding a fire in the emergency diesel generator (EDG) corridor potentially rendering all adjacent EDGs inoperable. The inspectors concluded that the licensee failed to update the UFSAR to reflect an adequate safe shutdown path for Fire Zone 5C11 from 1987 to July 1996 as required by 10 CFR 50.71(e).

Report Details

Exercise of Discretion

Three violations described in Section E8.1, E8.2, and E8.4 of this report are based upon licensee activities which were identified after, but occurred prior to the licensee announcing, on December 1996, an extended shutdown of the LaSalle County Station. These violations satisfy the appropriate criteria in Section VII.B.2, "Violations Identified During Extended Shutdowns or Work Stoppages," of the "General Statement of Policy and Procedures for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600, and Notices of Violation (NOVs) are not being issued for these violations. In particular, although the violations were not identified by the licensee, the other criteria specified in Section VII.B.2 of NUREG-1600 were met, which allows enforcement discretion to be applied. Specifically, in reference to the three violations, enforcement action was not considered necessary to achieve remedial action, the violations would not be categorized at Severity Level I, and the violations were not willful. In addition, actions specified in Confirmatory Action Letter RIII-96-008B effectively prevent the licensee from starting up LaSalle County Station without implicit NRC approval.

III. Engineering

E8 Miscellaneous Engineering Issues

E8.1 (Closed) Unresolved Item 50-373/374/96011-12

a. Inspection Scope

The inspector reviewed Unresolved Item 50-373/374/96011-12 regarding emergency core cooling system (ECCS) room cooler (VY) flow concerns.

b. Observations and Findings

Background

As discussed in NRC inspection report 50-373/374/96011, the inspectors reviewed portions of the pre-operational tests for the residual heat removal service water (RHRSW) system, the diesel generator cooling water (DGCW) system, and the ECCS room coolers (VY coolers). Following that review the following issues were identified:

- Pre-operational testing failed to identify a potentially significant interaction between the RHRSW and DGCW systems. Specifically, RHRSW backpressure effects on DGCW pump flow to the VY coolers had not been considered or tested although this condition would exist during a design basis event. As a result, the inspectors surmised that backpressure from RHRSW could adversely affect the flow through the VY coolers.

- In response to concerns regarding potentially excessive flow through the Unit 1 low pressure core spray (LPCS) room cooler (1VY04A), the licensee performed cooler flow velocity calculation VY-12, "Evaluation of VY Cooler Tube Velocity-Based on Test Data," Revision 0, dated September 13, 1996, which calculated a cooler flow that was half the value calculated by the inspectors. The inspectors noted that the licensee's calculation treated the 1VY004A cooler as two separate coolers, with half the flow going to each "sub" cooler. The basis for treating the cooler as two separate coolers was not understood at the time.

Licensee Response

The licensee responded to the inspectors' concerns described above in a letter dated December 20, 1996. In that response, the licensee provided the following information:

RHRSW Backpressure Concerns

Calculation L-000679, "Determination of Flow Correction Factors for Evaluating the Performance of Core Standby Cooling System - Equipment Cooling Service Water (CSCS-ECWS) Pump Operation," Revision 1, dated November 1, 1996, was performed to address the effects of operation of the RHRSW system on VY cooler performance. That calculation determined that adequate flow was provided to the VY coolers.

Cooler Flow Velocity Concerns

The 1(2)VY04A coolers have the same physical dimensions as other ECCS room coolers. However, since the 1(2)VY04A coolers are of a double serpentine coil arrangement, they have twice the number of cooling water flow circuits as the other coolers which only have a full serpentine coil arrangement. As a result, calculation VY-12 accurately models the VY cooler configuration.

Inspector Review

The inspector reviewed the licensee's response to the issues discussed above. The following was identified:

RHRSW Backpressure Concerns

The inspector reviewed calculation L-000679 and determined that in addition to evaluating RHRSW system discharge pressure effects, other factors such as lake level changes, strainer fouling, instrument inaccuracies, and suction pressure effects were also considered.

The inspector also identified that the calculation results determined that acceptable flow was not provided to the high pressure core spray (HPCS) diesel engine coolers, the Unit 2 division 2 VY coolers, and backwash flow for the Unit 1 and Unit 2 HPCS, and Unit 2 division 2 core standby cooling system (CSCS) strainers. However, following additional engineering evaluation, the calculated flow to these components was determined to be satisfactory. The inspector reviewed the licensee's calculations, including the additional engineering evaluations performed, and concluded that all CSCS components would receive adequate flow although the licensee had failed to appropriately consider the important factors noted above during pre-operational testing.

The inspector also determined that the licensee's consideration of suction pressure effects concluded that as long as three service water pumps were verified to be operating (48,000 gpm), the flow rate and therefore the friction losses in the suction piping and CSCS intake tunnel during testing would be greater than design conditions (37,100 gpm) and no correction to the surveillance testing acceptance criteria was necessary. However, the inspector identified that the applicable surveillance procedure, LOS-RH-Q1, "RHR (LPCI) and RHR Service Water Pump and Valve Inservice Test for Operational Conditions 1, 2, 3, 4 and 5," did not contain prerequisites which required that three service water pumps be operating. The inspector included that this was a weakness which could potentially result in inadequate testing initial conditions. The inspectors discussed this information with licensee personnel who stated that the procedure would be revised to ensure that three service water pumps were in service prior to testing.

10 CFR 50, Appendix B, Criterion XI, "Test Control," requires that a test program be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. The test program shall include, as appropriate, proof tests prior to installation, pre-operational tests, and operational tests during nuclear power plant operation.

As discussed in inspection report 50-373/374/96011, and as described above, pre-operational tests for ECCS room cooler flow measurement failed to incorporate the effects of RHRSW system operation on testing results as required by 10 CFR 50, Appendix B, Criterion XI. This issue was considered a violation. However, because this violation was based upon activities of the licensee prior to the events leading to the extended plant shutdown and satisfied the criteria in Section VII.B.2, "Violations Identified During Extended Shutdowns or Work Stoppages," of the "General Statement of Policy and Procedures for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600, an NOV is not being issued (NCV 50-373/374/97004-01).

Cooler Flow Velocity Concerns

The inspector reviewed the licensee response, including vendor drawings of the subject coolers, and concluded that the cooler configuration was as described by the licensee. This issue is closed.

c. Conclusions

The inspector concluded that adequate flow was provided to the VY coolers and that there were no further concerns regarding excessive flow through the 1VY04A room cooler. The inspector also concluded that pre-operational tests regarding ECCS room cooler flow requirement verifications failed to incorporate the effects of RHRSW system operation. In addition, surveillance procedure LOS-RH-Q1 did not establish initial conditions to require that three service water pumps were running although this equipment configuration was necessary for adequate testing. The inspectors reviewed the licensee's immediate and planned corrective actions and have no further concerns. Unresolved Item 50-373/374/96011-12 is closed.

E8.2 (Closed) Unresolved Item 50-373/374/96011-16

a. Inspection Scope

The inspector reviewed Unresolved Item 50-373/374/96011-16 regarding the effect of lake level on surveillance testing acceptance criteria.

b. Observations and Findings

Background

As discussed in NRC inspection report 50-373/374/96011, the inspectors identified that surveillance tests did not account for the effects of lake level on testing results. Specifically, the inspectors were concerned that although lake level was normally about 700 feet, surveillance testing did not adjust for the design basis lake level of 690 feet.

For example, LOS-RH-Q1, "RHR (LPCI) and RHR Service Water Pump and Valve Inservice Test for Operational Conditions 1, 2, 3, 4 and 5," which verified that the design flow of 7400 gallons per minute (gpm) through the residual heat removal (RHR) heat exchanger was met, may not be adequate since acceptance criteria established in LOS-RH-Q1 resulted in a condition in which RHRSW flow could be unacceptable under design basis conditions, although surveillance testing requirements were met due to the higher normal lake level.

Licensee Response

The licensee responded to the inspector's concerns described above in a letter dated December 20, 1996. In that response, the licensee provided the following information:

- Correction factors for lake level had not been included in surveillance procedures as appropriate.
- Problem Identification Form (PIF) 96-5484 was generated to identify the problem.
- Planned corrective actions included a review of all applicable surveillance procedures as part of the ongoing System Functional Performance Review (SFPR) program to ensure that these procedures included adequate prerequisite requirements and acceptance criteria.

In addition, the inspector determined that a computer model for CSCS system flow was being generated to establish appropriate RHRSW flow acceptance criteria.

Inspector Review

The inspector reviewed PIF 96-5484 which discussed the effects of lake level elevation differences on the acceptance criteria of LOS-RH-Q1 and noted the following:

- Based on preliminary calculations, the correction factor to account for design lake level was about 3.5 percent, which translated to about 260 gpm of RHRSW flow.

- This would result in a new minimum RHRSW flow acceptance criteria of 7660 gpm.

- A review of previous surveillance data indicated that flow had been above 7700 gpm for at least the last year.

The inspector reviewed Unit 1 and Unit 2 RHRSW test surveillance LOS-RH-Q1 results and confirmed that flow had been measured greater than 7700 gpm for the last year. However, the inspector also noted that recorded 1C/1D RHRSW pump flow was 7400 gpm for tests conducted on December 27, 1993, May 12, 1994, and December 27, 1994; and recorded 2C/2D RHRSW pump flow was 7600 gpm on January 10, 1995. All of these measured values were less than the revised value noted above. The inspector discussed this issue with licensee personnel who stated that some operators log flow at the acceptance criteria value although the actual values are much greater. The inspector reviewed this information against testing results and determined that since recorded flow was typically much greater than 7400, it was unlikely that the large flow variations as logged would occur. The inspector concluded that the licensee's trending and analysis of the measured data was poor since system engineering failed to identify and resolve this concern previously.

10 CFR 50, Appendix B, Criterion XI, "Test Control," requires that a test program be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents.

As discussed in inspection report 50-373/374/96011, and as described above, the inspector concluded that surveillance test acceptance criteria regarding minimum RHRSW flow failed to account for lake level changes and was a violation of 10 CFR 50, Appendix B, Criterion XI. However, because this violation was based upon activities of the licensee prior to the events leading to the extended plant shutdown and satisfied the criteria in Section VII.B.2, "Violations Identified During Extended Shutdowns or Work Stoppages," of the "General Statement of Policy and Procedures for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600, an NOV is not being issued (NCV 50-373/374/97004-02).

c. Conclusions

The inspector concluded that surveillance testing acceptance criteria failed to account for design basis lake level. The inspectors reviewed the licensee's immediate and planned corrective actions and have no further concerns. Unresolved Item 50-373/374/96011-16 is closed.

The inspector concluded that the licensee's trending and analysis of RHRSW flow was poor since system engineering failed to adequately address large variations in logged RHRSW flow.

E8.3 (Open) Unresolved Item 50-373/374/96011-19

a. Inspection Scope

The inspector reviewed Unresolved Item 50-373/374/96011-19 regarding a residual heat removal (RHR) heat exchanger waterhammer concern during RHRSW system startup.

b. Observations and Findings

Background

As discussed in NRC inspection report 50-373/374/96011, the inspectors identified that the RHR heat exchanger tubes were at an elevation significantly higher than the design basis lake level of 690 feet. As a result, the inspectors postulated that the heat exchangers could be susceptible to tube voiding as a result of an event which caused lake level to decrease from the normal level of 700 feet to the design level of 690 feet, or due to boiling in the tubes immediately following a design basis event. In either case, once the RHRSW system was started, the inspectors were concerned that a void in the tubes would rapidly collapse, resulting in a waterhammer which could rupture the RHR heat exchanger tubes.

The licensee responded during the inspection that a waterhammer would occur as postulated. At the end of the inspection, the licensee had not determined the effects of the postulated waterhammer.

Licensee Response

The licensee responded to the inspectors' concerns described above in a letter dated December 20, 1996. In that response, the licensee provided the following information:

Division 1

Due to the physical configuration of the Unit 1 and Unit 2 division 1 RHRSW systems, the only location within the system where the postulated waterhammer event could occur, under certain circumstances, was in the upper elevation of the RHR heat exchanger tubes. For example, if the RHR heat exchanger tubes were heated prior to starting the RHRSW pumps, such as during a LOCA or SRV blowdown, voiding could occur. In addition, the situation could be worsened if the lake level dropped 10 feet following a failure of the nonsafety-related dike.

As a result, the licensee conducted the following analyses:

- General Electric (GE) performed an analysis in October 1996 to develop conservative waterhammer pressure pulses resulting from a worst case waterhammer in the RHR heat exchanger tubes. GE concluded that the bounding waterhammer pressures which could occur in the RHR heat exchangers were acceptable.

- Sargent and Lundy (S&L) performed an analysis which provided further confirmation of the adequacy of the RHR heat exchangers following a postulated waterhammer in the heat exchanger tubes.

- S&L performed evaluations of the piping, equipment, and supports of the RHRSW system as a result of the conservative waterhammer event in the RHR heat exchanger tubes. This evaluation concluded that the stresses in the system piping, valves, penetrations, strainers, pumps, and supports were acceptable.

Division 2

The physical configuration of the Unit 1 and Unit 2 division 2 RHRSW system also allowed for the potential waterhammer event to occur in the heat exchanger tubes similar to division 1. However, the division 2 piping also contained a loop in which the piping has a large vertical rise and drop before terminating at the RHR heat exchanger inlet nozzle. This loop existed to provide a straight run of piping necessary to ensure the accuracy of flow measuring instrumentation.

The piping and piping supports could not be shown analytically to be able to withstand the postulated worst case waterhammer. As a result, a keep-fill system with cross-ties to both service water and diesel generator cooling water (DGCW) was designed for both division 2 RHRSW systems to maintain the piping system full and eliminate the potential for a waterhammer during design basis events.

Inspector Review

The inspector reviewed the licensee's response discussed above. The following issues were identified:

Division 1 Review

The inspector obtained the calculations performed by GE and S&L regarding evaluation of the stresses the RHR heat exchangers and RHRSW piping were expected to be subjected to in the waterhammer analysis. These calculations have been forwarded to NRR as a Task Interface Activity (TIA) for a technical review. Unresolved Item 93011-19 will remain open pending the results of that review.

Division 2 Review

The inspector reviewed the licensee response regarding the susceptibility of division 2 of RHRSW to waterhammer. In addition, the inspector determined that previous waterhammer events had occurred and had been documented back to May 1990. Further, the inspector determined that the effects of the waterhammer at normal lake level (700 feet) had been measured and procedure changes, including the installation of a temporary hose to provide makeup during planned system starts, had been implemented to minimize waterhammer effects. These actions appeared contrary to proper testing and corrective action practices. However, further inspector review was needed to determine the significance of the issues.

This is an Unresolved Item (50-373/374/97004-04) pending further NRC review.

c. Conclusions

The inspector determined that significant concerns regarding waterhammer on division 2 of the RHRSW system had been previously identified by the licensee due to a high vertical piping loop unique to division 2. As a result, the licensee planned to install a keep-fill modification for division 2.

In addition, the licensee evaluated the effects of potential RHR heat exchanger tube voiding and concluded that the stresses imposed on the RHRSW system due to a potential waterhammer were acceptable. The inspector obtained this evaluation and forwarded it to NRR for review.

E8.4 (Closed) Unresolved Item 50-373/374/96011-20

a. Inspection Scope

The inspectors reviewed Unresolved Item 50-373/374/96011-20 regarding a fire in the emergency diesel generator (EDG) corridor potentially rendering all adjacent EDGs inoperable.

b. Observations and Findings

Background

As discussed in NRC inspection report 50-373/374/96011, the NRC inspectors noted a licensee-identified issue open from 1987 to July 1996 related to a potential fire in an EDG corridor (Fire Zone 5C11) which could render all Unit 1 or Unit 2 EDGs inoperable. Specifically, a fire in the Unit 1 or Unit 2 EDG corridor could cause carbon dioxide fire protection system control panels for the adjacent diesel generator rooms to spuriously actuate. As a result, all EDGs adjacent to the affected corridor would be rendered inoperable due to inadequate ventilation although the licensee's fire hazards analysis contained in the Updated Final Safety Analysis Report (UFSAR) relied upon the EDGs for safe shutdown.

The action taken in 1987 was initiation of an hourly fire watch and a modification request to install physical protective barriers for the carbon dioxide fire protection system control panels. However, due to concerns with fire retardant materials (Thermo-lag) the modification package was put on hold in 1991 and was canceled in September 1996.

The licensee's basis for canceling the modification was the establishment of an alternative shutdown path; hot shutdown maintained by controlling vessel level and pressure with the reactor core isolation cooling (RCIC) system and the automatic depressurization system (ADS), which did not require EDG operation; and long-term actions, such as suppression pool cooling, accommodated by cross-tieing emergency buses to the other unit's system auxiliary transformer (SAT) or EDGs. The inspector determined that this alternative safe shutdown path was acceptable.

The inspectors questioned the licensee regarding the adequacy of the compensatory actions in place from 1987 to 1996 and the guidance available to operators in the event of a fire during this nine-year period. In addition, the inspectors requested that the licensee provide evidence to support an assertion that a fire in the EDG corridor would not result in a loss of normal power to affected components.

Licensee Response

The licensee responded to the inspectors' concerns described above in a letter dated December 20, 1996. In that response, the licensee provided the following information:

- UFSAR section H.4.2.57, "Safe Shutdown Analysis for Fire Zone 5C11," and associated tables were revised to add a new safe shutdown path in the event of a fire in Fire Zone 5C11 (EDG corridor) to satisfy the requirements of 10 CFR 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979."

- Procedures had been in place for operations to establish unit cross-ties to either unit.

  - Prior to 1987, LaSalle Abnormal Operating Procedure (LOA) LOA-AP-07, "Loss of Auxiliary Electrical Power." and LOA-AP-08, "Total Loss of AC Power," were in place.

  - From 1987, LOA-AP-101, "Unit 1, AC Power System Abnormal," and LOA-AP-201, "Unit 2, AC Power System Abnormal," were in place.

- Normal offsite power was supplied from the SATs to the safety-related 4.16 kV buses. In addition, although the bus ducts connecting the divisional busses to the SATs were routed in each unit's respective EDG corridor, a fire in these areas would not prevent these electrical connections from performing their functions (supplying power from the SATs to the divisional buses) for the following reasons:

  - The entire bus assembly consisted of non-combustible materials.

- The diesel generator corridors are provided with fire detection and automatic suppression water systems. In the event of a fire, the suppression system would operate to minimize exposure of the bus duct to flames. In addition, the bus duct enclosures are made of drip-proof, non-ventilated construction. Therefore, the bus assembly would not be exposed to any direct water sprays or internal water accumulation, and would not be expected to fail during any operation of any fire suppression systems.

- The primary combustibles in the diesel generator corridors consist of electrical cable insulation routed in solid bottom and open top cable trays. In addition, the division 2 power and control cable trays in each EDG corridor are provided with a Darmatt Type KM1, 1-hour rated fire barrier. Therefore, the cables routed in these trays are protected from fires and will not initiate or support a fire. Regarding the division 1 cables, while none of the cables in these trays are provided with a fire barrier, the cables in these trays are qualified to meet the Vertical Burner Flame Test in accordance with IEEE 383-1974. Therefore, an electrically initiated fire in these exposed cable trays will not propagate.

- The impact of a fire in an exposed cable tray relative to the bus ducts is minimized by the physical location of the exposed cable trays to the bus ducts, in addition to the automatic detection and suppression systems.

Inspector Review

The inspectors reviewed the response discussed above and questioned the adequacy of the justification that the bus ducts supplying power from the SATs to the 4.16 kV buses would not be affected by a fire in the EDG corridor. Following those discussions, the licensee determined that this justification was inadequate and issued a supplemental response to Unresolved Item 96011-20 in a letter dated February 24, 1997.

In that supplemental response, the licensee assumed that a fire in the Unit 1 and Unit 2 EDG corridor would affect the bus ducts supplying power from the respective unit SAT to the safety-related 4.16 kV buses.

In addition, the licensee documented that safe shutdown from the control room could be achieved utilizing RCIC, ADS, and RHR, in accordance with written procedures as follows:

- Hot shutdown achieved by automatic scram.
- Hot shutdown maintained by controlling vessel level and pressure with RCIC and ADS.
- Alternating Current (AC) power to equipment necessary for achieving and maintaining cold shutdown established by crosstieing the opposite unit SAT or EDG in accordance with LOA-AP-101 (Unit 1) or LOA-AP-201 (Unit 2).
- Cold shutdown maintained and achieved with RHR.

The inspectors reviewed the supplemental response and concluded that the safe shutdown path was acceptable. The inspectors also concluded that this safe shutdown path, although not formally documented in the safe shutdown analysis, was available to operators prior to identification of the issue in 1987.

The inspectors reviewed the licensee's actions in response to the identification of the problem in 1987 against 10 CFR 50, Appendix R requirements. 10 CFR 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," paragraph III.G.2 requires that except as provided for in paragraph III.G.3, where cables or equipment, including associated nonsafety-related 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:

- Separation of cables and equipment and associated nonsafety-related circuits of redundant trains by a fire barrier having a 3-hour rating; or

- Separation of cables and equipment and associated nonsafety-related circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards. In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area; or

- Enclosure of cable and equipment and associated nonsafety-related circuits of one redundant train in a fire barrier having a 1 hour rating. In addition, fire detectors and an automatic fire suppression system shall be installed in the area.

As identified by the licensee in 1987 and discussed above, Fire Zone 5C11 failed to meet the requirements of paragraph III.G.2. Specifically, the licensee identified in 1987 that a fire in the Unit 1 or Unit 2 diesel generator corridor could render the EDGs adjacent to the corridor inoperable due to potential failure of diesel generator room fire suppression systems located in the corridor, although the EDGs were required for safe shutdown as described in the UFSAR.

Paragraph III.G.3 of 10 CFR 50, Appendix R, requires, in part, that alternative or dedicated shutdown capability and its associated circuits, independent of cables, systems, or components in the area, room, or zone under consideration, shall be provided where the protection systems whose function is required for hot shutdown does not satisfy the requirement of paragraph III.G.2.

Although not formally documented or recognized by the licensee until July 1996, the alternative shutdown capability requirements of paragraph III.G.3 were met since an alternative shutdown path existed. However, the inspectors determined that the criteria specified by paragraph III.G.3 had not been satisfied until the licensee updated the UFSAR to identify an adequate alternative safe shutdown path. This was completed in July 1996. 10 CFR 50.71(e) required, in part, that the UFSAR be periodically updated to include the latest information developed. However, from 1987 to July 1996, the licensee failed to appropriately update the

UFSAR following identification that in the event of a fire in the EDG corridor, the adjacent EDGs would be rendered inoperable which was a violation of 10 CFR 50.71(e). However, because this violation was based upon activities of the licensee prior to the events leading to the extended plant shutdown and satisfied the criteria in Section VII.B.2, "Violations Identified During Extended Shutdowns or Work Stoppages," of the "General Statement of Policy and Procedures for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600, an NOV is not being issued (NCV 50-373/374/97004-03).

During the review of the UFSAR, the inspectors identified a discrepancy. Section H.4.3 of the UFSAR Safe Shutdown Analysis indicated that power to the RCIC room cooler fans must be provided within one hour to prevent exceeding RCIC qualification temperatures. However, in the station blackout analysis performed in 1990, the licensee determined that RCIC could operate for 4 hours without room cooling. Therefore, the UFSAR statement appeared to be different. Subsequently, the licensee generated a PIF to document this discrepancy.

c. Conclusions

The inspectors determined that although the licensee identified in 1987 that the fire protection requirements of Appendix R, paragraph III.G.2 were not met, the licensee did not recognize that the fire protection requirements of paragraph III.G.3 were met until July 1996 when an alternative shutdown path was identified. As a result, the inspectors concluded that the licensee failed to update the UFSAR to reflect an adequate safe shutdown path for Fire Zone 5C11 from 1987 to July 1996 as required by 10 CFR 50.71(e). The inspectors reviewed the immediate and planned corrective actions and have no further concerns. Unresolved Item 50-373/374/96011-20 is closed.

The inspectors concluded that UFSAR Section H.4.3 incorrectly stated that power to the RCIC room cooler fans must be restored within 1 hour although a 1990 station blackout analysis concluded that RCIC could operate up to 4 hours without room cooling.

V. Management Meetings

X1 Exit Meeting Summary

The inspector presented the results of their inspection activities to licensee management at an exit meeting on May 16, 1997. The licensee acknowledged the findings presented. The inspector asked the licensee if any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

P. Barnes, Regulatory Assurance
J. Drago, Regulatory Assurance
R. Gremchuk, System Engineering
T. Hammerich, System Engineering
A. Javorik, System Engineering Supervisor
D. Roberts, System Engineering
J. Rommel, Design Engineering
C. Snyder, System Engineering

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 64100:	Post-fire Safe Shutdown, Emergency Lighting and Oil Collection Capability at Operating and Near-Term Operating Reactor Facilities
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 92701:	Followup

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-373/374-97004-04	URI	RHRSW Division 2 Waterhammer

Closed

50-373/374/96011-12	URI	Determination of Maximum Flow Through 4A Room Cooler
50-373/374/96011-16	URI	Determination of Effect of Lake Level on RHRSW Flow
50-373/374/96011-20	URI	Potential Fire in EDG Corridor
50-373/374/97004-01	NCV	Determination of Maximum Flow Through 4A Room Cooler
50-373/374/97004-02	NCV	Determination of Effect of Lake Level on RHRSW Flow
59-373/374/97004-03	NCV	Potential Fire in EDG Corridor

Discussed

50-373/374/96011-19	URI	Determination of Effect of WaterHammer on RHR Heat Exchanger

LIST OF ACRONYMS USED

AC	Alternating Current
ADS	Automatic Depressurization System
AP	Auxiliary Power
CFR	Code of Federal Regulations
CSCS	Core Standby Cooling System
DGCW	Diesel Generator Cooling Water
ECCS	Emergency Core Cooling System
ECWS	Equipment Cooling Service Water
EDG	Emergency Diesel Generator
fps	Feet Per Second
GE	General Electric
gpm	Gallons Per Minute
HPCS	High Pressure Core Spray
IEEE	Institute of Electrical and Electronics Engineers
IP	Inspection Procedure
kV	Kilovolt
LOA	LaSalle Abnormal Operating Procedure
LOS	LaSalle Operating Surveillance
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LST	LaSalle Special Test
NFPA	National Fire Protection Association
NOV	Notice of Violation
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
PDR	Public Document Room
PIF	Problem Identification Form
PT	Performance Test
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
SAT	System Auxiliary Transformer
SFPR	System Functional Performance Review
S&L	Sargent and Lundy
TIA	Task Interface Activity
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VIO	Violation
VY	Emergency Core Cooling System Room Cooler