James A. FitzPatrick
Nuclear Power Plant
P.O. Box 41
Lycoming, New York 13093
315 342-3840

New York Power Authority

Harry P. [illegible], Jr.
Resident [illegible]

April 20, 1992
JAFP-92-0329

United States Nuclear Regulatory Commission
Document Control Desk
Mail Station P1-137
Washington, D.C. 20555

SUBJECT: DOCKET NO. 50-333
LICENSEE EVENT REPORT: 92-015-00 - Appendix R Reanalysis Reveals Safe Shutdown Analysis Deficiencies

Dear Sir:

This report is submitted in accordance with 10 CFR 50.73(a)(2)(ii) and (v).

Questions concerning this report may be addressed to Mr. W. Verne Childs at (315) 349-6071.

Very truly yours,

HARRY P. SALMON, JR.

HPS:WVC:lar

Enclosure

cc: USNRC, Region I
USNRC Resident Inspector
INPO Records Center

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NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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TITLE (4) Appendix R Reanalysis Reveals Safe Shutdown Analysis Deficiencies

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 3	1 7	9 2	9 2	— 0 1 5 —	0 0	0 4	2 0	9 2		0 5 0 0 0
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THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 0

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	X 50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	X 50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
W. Verne Childs, Senior Licensing Engineer	AREA CODE 3 1 5 3 4 9 - 6 0 7 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) X NO

EXPECTED SUBMISSION DATE (15) MONTH DAY YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

EIIS Codes are in []

The plant was shutdown and in the cold condition for maintenance and refuel. On 3/17/92, as a result of reanalysis of the 1982 and 1985 10 CFR 50, Appendix R analyses, seven deficiencies with the potential to effect safe shutdown capability during postulated fires were identified. Root causes of the deficiencies were identified as a lack of management commitment to the fire protection program implementation, inadequate interface between the headquarters and plant staffs, and inadequate staff qualification in the area of fire protection. Corrective actions include correction of the deficiencies, plant and headquarters reorganization, improvements in staff qualification, and additional staff. LER-91-010, 91-021, and 91-023 describe additional fire protection deficiencies.

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Description

The plant was shutdown and in the cold condition for maintenance and refuel.

A reanalysis of 10 CFR 50, Appendix R, requirements was initiated in August 1991. The contractor performing the reanalysis provided the New York Power Authority (NYPA) with a preliminary report of the results of the reanalysis on February 24, 1992. On March 17, 1992, based on review of the preliminary report, seven (7) deficiencies were identified as conditions which potentially require reporting under 10 CFR 50.72 and 10 CFR 50.73. The NRC was notified of the seven deficiencies on March 17, 1992 at 1535 hours via the Emergency Notification System (ENS). On March 19, 1992 the Plant Operations Review Committee reviewed the deficiencies and determined they were conditions requiring a report under 10 CFR 50.73 (LER system).

It should be noted that this LER is based on the preliminary reanalysis report. The report has not been finalized. Review and acceptance of the report in accordance with design control procedures has not been completed. Additional deficiencies may be identified as part of the continuing review process.

Each of the seven deficiencies identified is described below.

1. Residual Heat Removal/Low Pressure Coolant Injection (RHR/LPCI) [BO] heat exchanger bypass valve 10MOV-66A and injection valves 10MOV-25A and -27A may not be operable as a result of postulated fires within the reactor building [NG] Fire Areas VIII and IX.

   Power to valves 10MOV-66A, 10MOV-25A, and 10MOV-27A is provided from safety-related 600 VAC [ED] Motor Control Center (MCC) 155. The normal power feeder cable for MCC-155 is routed through Fire Areas VIII and IX. As a result, it is assumed that postulated fires in Fire Areas VIII and/or IX would result in loss of MCC-155 due to damage to the feeder cable.

   While Abnormal Operating Procedures (AOPs) address power restoration to MCC-155 in response to loss of the normal feed, portions of the feeder cable which are routed through Fire Areas VIII and IX could be damaged in such a manner that it would not be possible to restore power to MCC-155 because the damaged (faulted) feeder cable is not isolated from MCC-155. In other words, tripping of the normal feeder breaker for MCC-155 would not isolate fire-induced MCC-155 feeder cable faults to allow power restoration from the alternate feeder. As a result, postulated Fire Area VIII and/or Fire Area IX fires could cause loss of power to RHR/LPCI Loop A heat exchanger bypass valve 10MOV-66A. When power is not available for 10MOV-25A and -27A, core cooling water make-up for boil-off and/or for reflooding would not be available. When power is not available for 10MOV-66A, long-term reactor core [AC] and primary containment [NH] pressure suppression pool (torus) cooling would not be available from RHR/LPCI Loop A because valve 10MOV-66A is normally open. No credit is taken for local manual positioning of the valves because access cannot be assured for all postulated fires.

2. A postulated control room [NA] or relay room [NA] fire could damage the control circuits or actuation logic for Safety Relief Valves (SRVs) or the Automatic Depressurization System (ADS) and thus could cause the opening of one or more valves resulting in the rapid reactor depressurization and inventory loss. Low pressure emergency core cooling systems (RHR/LPCI and/or core spray [BM]) cannot be assumed to be immediately operable as a result of damage due to the postulated control room or relay room fire.

3. A postulated fire in Fire Area VIII of the reactor building [NG] may damage cables that result in loss of reactor pressure indication in the control room [NA]. The potentially effected instruments are 06PI-61A, 06PI-61B, 06PR-61A, 06PR-61B, 06PI-62, 06PI-90A, 06PI-90B, and 06PI-90C.

   Pressure indicators 06PI-61A and -61B and pressure recorders 06PR-61A and -61B are accident monitoring [IP] instruments on control room Panel 09-3. Pressure indicator 06PI-62 provides reactor pressure indication adjacent to Safety Relief Valve (SRV) and ADS controls on control room Panel 09-4. Pressure indicators 06PI-90A, -90B, and -90C on control room panel 09-5 provide the normal indication of pressure used by operators and are also associated with the reactor feedwater control system [JB].

4. Reactor vessel head vent valves 02AOV-17 and -18 were assumed to be maintained closed in the 1982 and 1985 10 CFR 50, Appendix R, analyses. The 1982 and 1985 analyses indicated (in error) that adequate physical separation for the valve control circuits existed. Reevaluation has shown that control circuits for solenoid valves 02SOV-17 and -18 (which control the pneumatic supply to 02AOV-17 and -18) do not have adequate physical separation to provide assurance that a postulated reactor building fire would not effect both reactor head vent valves.

The physical separation is inadequate because control circuits for both valves penetrate the primary containment [NH] drywell through the same electrical penetration assembly.

Spurious opening of both reactor head vent valves due to postulated fire damage to the control circuits would result in a small reactor vessel steam leak similar to a small break loss of coolant accident.

5. Reanalysis identified a number of Safety Division 1 cables potentially damaged as a result of a postulated fire in Fire Area IA. The 1982 and 1985 analyses indicated that safe shutdown could be achieved using Safety Division 1 equipment for a postulated fire in Area IA.

   Equipment potentially effected by postulated fires in Area IA include the following:

   - Division 1 reactor water level transmitters 02-3LT-72A and -72C
   - Division 1 reactor pressure transmitters 02-3PT-52A and -52C
   - Division 1 reactor level indicators 02-3LI-85A, -91, and -92
   - Division 1 RHR/LPCI flow indication
   - Division 1 RHR service water flow indication
   - Several Division 1 RHR/LPCI valve control circuits
   - Division 1 RHR service water pump control circuits
   - Division 1 RHR pump 10P-3A control circuits

6. The 1985 analysis indicated that power would be available to outboard main steam [SB] line drain valve 29MOV-77 and outboard High Pressure Coolant Injection (HPCI) [BJ] system steam line isolation valve 23MOV-60 to complete the primary containment [NH] isolation function for postulated cable spreading room [NA] fires. Reanalysis identified an error in the 1985 analysis. A 125 VDC [EJ] power cable in the cable spreading room could be damaged resulting in loss of power to valve 23MOV-60 and 29MOV-77.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

7. The 1985 analysis relies on Emergency Service Water (ESW) [BI] pump 46P-2B to support safe shutdown for postulated control room [NA], relay room [NA], and/or cable spreading room [NA] fires. Controls for ESW pump 46P-2B are provided at the alternate shutdown panel. Reanalysis indicates that the pump may not be available because the ESW pump isolation switch on the alternate shutdown panel 25ASP-3 does not completely isolate the control circuit from the control room, relay room, and cable spreading room.

Additional fire protection deficiencies have been identified and are discussed in NRC Inspection 92-80.

Cause

An analysis was performed to determine the root causes of the fire protection organizational and programmatic problems. The analyses indicates the following primary causes:

1. Lack of a commitment to fire protection program implementation by NYPA management.

2. Inadequate interface between headquarters and plant staff organizations.

3. Inadequate staff qualification in the fire protection area.

Analysis

Each of the seven deficiencies results in conditions in which safe shutdown, as a result of postulated fire conditions, cannot be assured as required by 10 CFR 50, Appendix R. Accordingly, the event requires a report under 10 CFR 50.73(a)(2)(ii)(B); that is, the events result in conditions which are outside of the design basis.

The event also requires a report under 10 CFR 50.73(a)(2)(v)(B) and (C); that is, conditions that could have prevented the fulfillment of the safety function of systems needed to remove residual heat and control the release of radioactive material.

Corrective Action

1. Modifications will be completed prior to start-up, or justification for the existing condition (including appropriate compensatory action) will be completed prior to start-up, following the 1992 Refuel Outage.

2. Corrections for the root causes are being added to the NYPA Nuclear Generation Business Plan and/or in the FitzPatrick Plant Results Improvement Plan.

Additional Information

Failed Components: None

Similar Events: LER-91-010, 91-021, and 91-023 reported additional fire protection program deficiencies.