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New York Power Authority

Radford J. Converse
Resident Manager

March 6, 1992
JAFP-92-0195

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

SUBJECT: DOCKET NO. 50-333
LICENSEE EVENT REPORT: 92-009-00 - Accident Monitoring Instrumentation Seismic Qualification

Dear Sir:

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

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

Very truly yours,

RADFORD J. CONVERSE

RJC:WVC:lar

Enclosure

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

Cert NO P640096447
IE22
1/1

NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

# LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)
JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NUMBER (2): 0 5 0 0 0 3 3 3

PAGE (3): 1 OF 0 5

TITLE (4): Operating Experience Review Reveals Accident Monitoring Instrumentation Configuration Different Than Seismic Qualified Configuration

| 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 NAME | DOCKET NUMBER (8) |
| 0 2 | 0 5 | 9 2 | 9 2 | — 0 0 9 | — 0 0 | 0 3 | 0 6 | 9 2 | | 0 5 0 0 0 |
| | | | | | | | | | | 0 5 0 0 0 |

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

| | | | |
|---|---|---|---|
| 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) | X 50.73(a)(2)(i) | 50.73(a)(2)(viii)(A) | |
| 20.405(a)(1)(iv) | 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: W. VERNE CHILDS, SENIOR LICENSING ENGINEER

TELEPHONE NUMBER — 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)

X YES (If yes, complete EXPECTED SUBMISSION DATE)

NO

EXPECTED SUBMISSION DATE (15): MONTH 0 7 — DAY 3 0 — YEAR 9 2

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

INTERIM REPORT

EIIS Codes are in []

The plant was shutdown and in the cold condition for maintenance and refuel. On 2/5/92 during an operating experience review of Information Notice 91-70 it was discovered that certain accident monitoring [IP] and Safety Parameter Display System Data input instrumentation that was installed in 1982 did not include installation of dummy modules. As a result, the installed configuration is different than the configuration used for seismic qualification of the equipment panel. The equipment of concern was administratively declared inoperable on 2/12/92 when it was understood that the condition requires reports under 10 CFR 50.72 and 10 CFR 50.73. The equipment was provided by an architect-engineer (A-E) without instructions related to installation of dummy modules, vibration dampeners, or similar parts needed to maintain operability during postulated seismic events. Discovery of additional deficiencies may result from internal inspections which have not yet been completed Dummy modules and any other parts needed to bring the equipment configuration into agreement with the qualified configuration will be installed prior to start-up. The A-E will be requested to determine the cause of the missing instructions.

NRC Form 366
(9-83)

INTERIM REPORT

EIIS Codes are in []

Description

The plant was shutdown and in the cold condition for maintenance and refuel. On February 5, 1992, as a result of an operating experience review, ten safety-related instrument channels were administratively declared inoperable because the installed physical configuration was different than the configuration used for seismic qualification of the equipment.

NRC Information Notice 91-70, dated November 4, 1991, discussed the discovery of improper installation of instrumentation modules in Foxboro Spec 200 instrument nests and modules at another facility. The improper installation involved vibration dampening materials, instrument guide rails, and brackets which were not installed. As a result, the equipment could fail during postulated seismic events.

During an operating experience review, which was initiated following receipt of Information Notice 91-70, instrumentation installed in April 1982 to provide the accident monitoring instrumentation [IP] to meet the requirements of Regulatory Guide 1.97 and provide input to the Safety Parameter Display System (SPDS) was found with deficiencies similar to those discussed in the Information Notice. Specifically, Panel 27MAP (monitoring and analysis panel) has vacant (empty) Safety Division 1 and 2 instrument nests. Dummy modules should have been installed in the vacant nests to maintain the equipment configuration consistent with the seismic qualified configuration. The absence of the dummy modules could result in instrumentation failures under postulated seismic conditions. A complete internal inspection of Panel 27MAP has not yet been performed to verify the presence of the required vibration dampeners, guide strips, and simila parts.

The instrumentation channels potentially effected provide control room [NA] indication and recorder displays as well as SPDS input data for the following parameters.

1. Primary containment [NH] pressure suppression chamber (torus) water level Channels A and B.

2. Primary containment drywell level Channels A and B.

3. Narrow-range drywell pressure Channels A and B.

4. Wide-range drywell pressure Channels A and B.

5. Reactor pressure Channels A and B.

On February 12, 1992 it was determined that the deficiencies resulted in a condition requiring a report under 10 CFR 50.72 and 10 CFR 50.73. The NRC was informed via the Emergency Notification System (ENS) on February 12, 1992 at 1315 hours.

The instruments are also required by Technical Specification Table 3.2-8 to be operable when the reactor is critical and/or when reactor temperature is equal to or greater than 212 F. Since the dummy modules were not installed when the accident monitoring instrumentation was installed, the instruments are considered to have been inoperable since installation in April 1982. The actions required by Table 3.2-8 for inoperable accident monitoring instruments were not taken because the condition which resulted in the declaration of inoperability was not discovered until February 5, 1992. No action was required at the time the condition was discovered because the plant has been shutdown and cooled down since November 28, 1991.

Panel 27MAP was installed as part of a modification. The architect-engineer (A-E) for the modification project used equipment provided by Foxboro. Panel 27MAP was provided to the plant as a completely engineered unit which only required mounting, assembly of internals, and termination of external connections to complete the installation of the panel. The documentation provided by the A-E did not include any instructions, cautions, or other information with respect to installation of dummy modules, guide strips, vibration dampeners, or similar materials necessary to maintain the seismic qualification of the equipment.

Additional instrumentation provided by the same manufacturer is contained in Panels 09-24 and 09-25. These panels also contain portions of instrumentation channels for accident monitoring and SPDS data input from Residual Heat Removal Service Water (RHRSW) [BI], Emergency Service Water (ESW) [BI], High Pressure Coolant Injection (HPCI) [BJ], Core Spray [BM], Reactor Core Isolation Cooling (RCIC) [BN], and Residual Heat Removal/Low Pressure Coolant Injection (RHR/LPCI) [BO].

Inspections of the internals of Panels 09-24 and 09-25 have not yet been completed. As a result, it is not known at this time if any of the deficiencies discussed in Information Notice 91-70 also exist in those panels. Additional deficiencies could potentially affect additional accident monitoring and/or SPDS data input channels.

Cause

The cause of the event has not been determined at this time. The equipment was provided as part of a plant modification. The modification architect-engineer (A-E) did not provide any instructions with respect to maintaining the seismic qualification of the equipment.

The A-E (General Electric) will be requested to conduct an investigation to determine the reason that no instructions with regard to maintaining seismic qualification were provided. This LER will be updated to provide the information.

Since complete inspection of the internals of Panels 27MAP, 09-24, and 09-25 have not been completed, it is not known whether or not similar additional conditions exist. The update of this LER will also include information with regard to the cause of any additional deficiencies found (if any).

Analysis

Accident monitoring instrumentation and the SPDS system provide indication of important plant parameters to operators during and following an accident. The information provided to operators is used to verify that plant response is consistent with the expected response for design basis accidents described in the Final Safety Analysis Report (FSAR). The information (along with information provided by other instrumentation) could also be used by operators during the decision processes associated with use of emergency operating procedures during and following both design basis and more severe accidents.

As a result, the event requires a report under 10 CFR 50.73(a)(2)(i)(B) due to plant operation prohibited by Technical Specifications; that is, the action required by Table 3.2-8 for inoperable instrumentation was not taken. In addition, the event is reportable under 10 CFR 50.73(a)(2)(v) as a condition that alone could have prevented the fulfillment of the safety function of systems needed to mitigate the consequences of an accident; that is, failure of the instruments could reduce the ability of the operator to take timely and/or proper actions to mitigate the consequences of an accident due to inadequate or erroneous information provided by the accident monitoring instrumentation and/or the SPDS system.

Corrective Action

1. No immediate corrective action was necessary because the plant was shutdown and in the cold condition. The accident monitoring instrumentation is not required to be operable when the reactor is shutdown and in the cold condition.

2. Dummy modules and any additional parts required will be installed in Panel 27MAP to make the installed configuration consistent with the seismic qualified configuration prior to plant start-up following the 1992 Refuel Outage. Scheduled due date May 14, 1992.

3. A complete inspection of Panels 27MAP 09-24, and 09-25 to verify that the installed configuration is consistent with the seismic qualified configuration and correction of additional deficiencies (if any) will be completed prior to plant start-up following the 1992 Refuel Outage. Scheduled due date May 14, 1992.

4. The architect-engineer (A-E) for the modification which installed Panel 27MAP in 1982 will be requested to provide (within 90 days) the results of an evaluation conducted to determine why the information with regard to maintaining the seismic qualification was not provided. The A-E will also be requested to provide information with respect to corrective actions taken to reduce the probability of recurrence. Due date for letter to A-E, March 25, 1992.

5. This LER will be updated within 30 days of the receipt of information from the A-E concerning the action requested in #4 above. Scheduled due date July 30, 1992.

Additional Information

Failed Components: None

Previous Similar Events: It cannot be determined if similar events have occurred at this facility until the cause(s) of the event are known. When this LER is updated, similar events (if any) will be noted.