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New York Power Authority

William Fernandez II
Resident Manager

October 23, 1989
JAFP-89-0759

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

REFERENCE: DOCKET NO. 50-333
LICENSEE EVENT REPORT: 88-008-01

Dear Sir:

Enclosed please find referenced Licensee Event Report in accordance with 10 CFR 50.73.

If there are any questions concerning this report, please contact Mr. W. Verne Childs at (315) 349-6071.

Very truly yours,

WILLIAM FERNANDEZ

WF:WVC:lar

Enclosure

cc: USNRC, Region I (1)
INPO Records Center, Atlanta, GA (1)
American Nuclear Insurers (1)
Internal Power Authority Distribution
NRC Resident Inspector
Document Control Center
LER/OR File



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

# LICENSEE EVENT REPORT (LER)

| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| JAMES A. FITZPATRICK NUCLEAR POWER PLANT | 0 5 0 0 0 3 3 3 | 1 OF 0 3 |

TITLE (4)
Excessive Leakage of Primary Containment Isolation Valves

| 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 9 | 0 1 | 8 8 | 8 8 | – 0 0 8 – | 0 1 | 1 0 | 2 3 | 8 9 | | 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) | [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 | |
|---|---|---|
| | AREA CODE | |
| W. VERNE CHILDS, SENIOR LICENSING ENGINEER | 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 |
|---|---|---|---|---|---|---|---|---|---|
| X | B J | I S V | P 3 0 5 | Y | | | | | |
| | | | | | | | | | |

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)

Update Report - Previous Report Date 9/21/88

EIIS codes are in []

During the 1988 refuel outage, one primary containment [NM] penetration exceeded the Technical Specification 4.7.A.2.b.(2) limit of 0.6 La (3,216 standard cubic feet per day) when subjected to Local Leak Rate Testing.

The leaking penetration valves were in the High Pressure Coolant Injection [BJ] turbine exhaust line. Leakage was attributed to wear.

Corrective action was to replace both leaking valves. Replacement was planned prior to the outage as part of a comprehensive leakage reduction program.

Safety significance and consequences are judged to be very small because the pathway of potential releases includes filters, dilution, delay, and elevated release from the plant stack [VL] for accidents discussed in the Final Safety Analysis Report.

LER-87-001, 85-008, 83-002, 81-078, and 80-050 are similar previous events.

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

Update Report - Previous Report Date September 21, 1988

EIIS codes are in []

Description and Cause of Event

During the 1988 refueling, maintenance, and modification outage, performance of Local Leak Rate Testing (LLRT) of primary containment [NM] penetration X-214 on September 1, 1988 revealed leakage in excess of Technical Specification 4.7.A.2.d.(2) limit of 0.6 La or 3,216 standard cubic feet per day (SCFD). Penetration X-214 has a configuration of two (2) check valves in series on the High Pressure Coolant Injection (HPCI) [BJ] turbine exhaust line to the primary containment suppression pool where low pressure turbine exhaust steam is condensed. As-found leakage of the inboard isolation valve (23-HPI-12) was approximately 3,736 SCFD, and the as-found leakage of the outboard valve (23-HPI-65) was approximately 7,014 SCFD. The cause of the deficiency is attributed to wear.

Analysis of Event

Leakage through HPCI turbine exhaust valves 23-HPI-12 and -65 would not be expected to be significant during either normal or accident conditions.

During normal plant operation conditions, the pressure suppression chamber is at or very near atmospheric pressure. As a result, little or no pressure differential exists to cause flow from the suppression chamber atmosphere through the valves. In addition, any leakage past the valves would be into closed piping within the secondary containment (reactor building) [NG]. The reactor building is normally maintained at a pressure slightly below atmospheric pressure, and the ventilation exhaust is continuously monitored for radioactive materials. If significant amounts of radioactive material are present in the ventilation exhaust, the normal reactor building ventilation system is automatically isolated and the building is then maintained at a slightly negative pressure by the Standby Gas Treatment (SGT) system [BH]. Automatic isolation of the normal reactor building ventilation and automatic starting of the SGT system also occur during accident conditions. Air flow through the SGT system is filtered by activated charcoal and High Efficiency Particulate Air (HEPA) filters prior to release from the elevated plant stack [VL]. Any effect on the off-site dose would be expected to be extremely small.

During a Loss of Coolant Accident (LOCA) in which the HPCI system is operating, leakage through the valves is not possible.

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

During a LOCA in which HPCI is not operating, leakage of radioactive materials from the pressure suppression chamber atmosphere would be expected to be higher than during normal operation due to the presence of higher pressure in the suppression chamber atmosphere as a result of the LOCA.

During a LOCA, radioactive materials would be "scrubbed" by passing through the suppression chamber water and (except for noble gases) most of the radioactive material would not be present in the suppression chamber atmosphere for leakage through valves 23-HPI-12 and -65. Any leakage of radioactive material from the closed HPCI turbine exhaust piping to the reactor building would also be diluted and delayed by the large volume of the reactor building. As a result of the scrubbing, dilution, delay, filtering, and elevated release, the effects on off-site dose would be very small.

Corrective Action

Plans for replacement of the valves had been made prior to the 1988 Refuel Outage plant shutdown as part of a comprehensive containment isolation valve replacement program. Both valves were replaced with new valves of a different design and were subjected to LLRT prior to plant startup. The "as-left" leakage rate for Penetration X-214 following valve replacement was 96.4555 SCFD.

Additional Information

Failed component identification:

Valves 23-HPI-12 and -65

Manufacturer - Wm. Powell Company

NPRD Code - P305

Model Number - 1561A-WE

Previous Similar Events:

LER-87-001, 85-008, 81-078, and 80-050 are previous events in which the valves for primary containment penetrations were found leaking in excess of 0.6 La.