NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

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 4

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 5 0 0 0
0 9	0 1	8 8	8 8	0 0 8	0 0	0 9	2 1	8 8		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
W. VERNE CHILDS, SENIOR LICENSING ENGINEER	AREA CODE 3 1 5 3 - 9 - 6 3 0 5

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	F 3 0 5	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

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

EXPECTED SUBMISSION DATE (15): MONTH 1 2, DAY 0 9, YEAR 8 8

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

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 are in the High Pressure Coolant Injection [BJ] turbine exhaust line. Leakage is attributed to wear.

Corrective action is 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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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.

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

Both valves are being replaced with new valves of a different design and will be subjected to LLRT prior to plant startup. Plans for replacement of the valves had been made prior to plant shutdown as part of a comprehensive containment isolation valve replacement program. Additional details of the valve replacement program are contained in References 1 and 2.

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.

An update of this report will be submitted approximately 30 days after the completion of the outage. The update will provide information concerning LLRT results for additional penetrations (if any) which are found with leakage greater than 0.6 La. The update will also provide a

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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JAMES A. FITZPATRICK NUCLEAR POWER PLANT	0 5 0 0 0 3 3 3	8 8	— 0 0 8	— 0 0	0 4 OF 0 4

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

summary of corrective action and after-repair LLRT results for those penetrations which were found with leakage in excess of 0.6 La.

References:

1. Letter from J. Brons (NYPA) to NRC dated April 8, 1988 (Serial Number JPN-88-012)

2. Letter from J. Brons (NYPA) to NRC dated June 17, 1988 (Serial Number JPN-88-031)

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

Radford J. Converse
Resident Manager

NewYork Power Authority

September 21, 1988
JAFP-88-0882

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

REFERENCE: DOCKET NO. 50-333
LICENSEE EVENT REPORT: 58-008-00

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) 349-6305.

Very truly yours,

RADFORD J. CONVERSE

RJC: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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