James A. FitzPatrick
Nuclear Power Plant
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New York Power Authority

Radford J. Converse
Resident Manager

February 28, 1992
JAFP-92-0179

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

SUBJECT: DOCKET NO. 50-333
LICENSEE EVENT REPORT: 92-008-00 - PCIV Stem Packing Not Subjected to LLRT

Dear Sir:

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

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

Very truly yours,

R. Losero by direction

RADFORD J. CONVERSE

RJC:WVC:lar

Enclosure

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

Cert No P064705453

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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 4

TITLE (4): Primary Containment Isolation Valve Stem Packing Not Subjected to Local Leak Rate Testing Following Maintenance on the 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 1	2 9	9 2	9 2	— 0 0 8 —	0 0	0 2	2 8	9 2		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)	[ ] 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	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	MANUFAC-TURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

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

EXPECTED SUBMISSION DATE (15): MONTH 0 6 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 1/29/92 it was determined that Local Leak Rate Testing (LLRT) of 2 primary containment [NH] isolation valves following maintenance, which may have changed the leakage rate of the valve stem packings, did not subject the valve stem packings to test pressure. Design and physical orientation of the valves prevents application of pressure on the valve stem packing during LLRT. The deficiency was discovered as part of review of NRC Information Notice 86-16 and was caused by a design error and inadequate (not timely) review of operating experience. Corrective actions have not been determined at this time. An update report will be submitted.
LER-92-005 described an event with similar causes.

NRC Form 366 (9-83)

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

Plans for correction of the design deficiency associated with LLRT capability of valves 27AOV-113 and -117 have not been finalized at the time this LER is submitted.

An updated report will be submitted by June 30, 1992 to provide specific corrective action information.

Additional Information

Failed Components: None

Previous Similar Events: LER-92-005 describes an event which involved an operating experience review deficiency and original installation design error. There have not been any other LERs concerning inadequate LLRT at this facility.

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

INTERIM REPORT

EIIS Codes are in []

Description

The plant was shutdown and in the cold condition for maintenance and refuel. On January 29, 1992 it was determined that the valve stem packing of two (2) primary containment [NH] isolation valves (27AOV-113 and 27AOV-117) had not been subjected to leakage testing after maintenance on the valves, that may have changed the valve stem packing leakage characteristics, was performed in August 1990.

The valves of concern are Fisher 9200 Series butterfly valves and are primary containment vent and purge system inboard isolation valves. The valve design and physical orientation is such that valve stem packing is on the primary containment side of the valve disc. As a result, when Local Leak Rate Testing (LLRT) of the valves is conducted to meet the requirements of 10 CFR 50, Appendix J, by pressurizing the volume contained between the inboard and outboard isolation valves, the valve stem packing is not subjected to test pressure.

Valves 27AOV-113 and 27AOV-117 were last subjected to leak testing as part of the most recent Primary Containment Integrated Leakage Rate Test (PCILRT) which was completed in June 1990. In August 1990, during a plant maintenance outage, the valve stem to valve disc pins in 27AOV-113 and -117 were replaced. While replacement of the pins did not require removal and replacement of the valve stem packing, the physical forces involved in removal of the old pins and installation of the new pins could result in changes in the packing leakage. Following completion of the pin replacement, the valves were subjected to LLRT as required. This LLRT was inadequate because, as noted above, the test does not subject the valve stem packing to test pressure. During the time period between repacking of the valve stems in August 1990 and discovery of the deficiency on January 29, 1992, the plant was operated for approximately 311 days.

Potential and actual LLRT program deficiencies were previously noted at other facilities by the NRC and the industry was informed by NRC Inspection and Enforcement Information Notice No. 86-16 issued on March 11, 1986. It was during formal operating experience review of Information Notice 86-16 that potential changes to the valve stem packing leakage on 27AOV-113 and -117 (without an adequate LLRT following the maintenance activities) was discovered. A total of seven (7) primary containment inboard isolation valves in the vent and purge system and the primary containment pressure suppression chamber (torus) to reactor building [NG] vacuum relief [BF] lines are designed and physically oriented in such a manner that the valve stem packing is not subjected to test pressure during LLRT. The valve stem packings are tested during PCILRT.

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

Cause

The event was caused by an original construction design error and by the lack of a timely operating experience review. Information Notice 86-16 discussed LLRT program deficiencies at three (3) boiling water reactor plants of approximately the same age and with designs very similar to the FitzPatrick plant. One of the deficiencies noted in the Information Notice concerned the inability (due to design and physical orientation) to subject the torus to reactor building [NG] vacuum relief line isolation valve stem packing to LLRT pressure. Precisely the same deficiency on a functionally identical system and isolation valves exists at the FitzPatrick plant. Review of the Information Notice under the operating experience review program was not prompt. The operating experience review of Information Notice 86-16 was not completed until more than five years after issue by the NRC.

Analysis

The primary containment is the primary barrier designed to withstand the pressures and temperatures resulting from a design basis Loss of Coolant Accident (LOCA) and provides hold-up for radioactive decay of any radioactive material released from the reactor coolant system pressure boundary. The leak tightness of primary containment is periodically demonstrated by PCILRT or, on a local basis for individual valves and seals, by LLRT. Since the valve stem leakage on valves 27AOV-113 and -117 could have been changed by the maintenance activity, and the subsequent LLRT did not test the potential leakage path at the valve stems, the ability of the packings to limit leakage is not known.

As a result, the event is reportable under 10 CFR 50.73(a)(2)(v)(C) and (D) as a condition that alone could prevent the fulfillment of the safety function of the primary containment that is needed to control the release of radioactive material and to mitigate the consequences of an accident.

Corrective Action

The Operating Experience Review Program was previously noted to contain weaknesses with regard to prioritization and/or timely review of significant review tasks. A comprehensive audit of operating experience, which was recently completed, has also identified a number of other programmatic deficiencies. The plant Results Improvement Program includes actions which will result in timely and thorough review of both internal (in-house) and external (industry-wide) operating experience.