James A. FitzPatrick
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NewYorkPower Authority

Michael J. Colomb
Site Executive Officer

January 5, 2000
JAFP-00-0003

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

Subject: **Docket No. 50-333**
LICENSEE EVENT REPORT: LER-99-013 (DER-99-02838)

Steam Leakage Detection System Outside Design Bases

Dear Sir:

This report is submitted in accordance with 10 CFR 50.73(a)(2)(ii)(B), "Any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being in a condition that was outside the design basis of the plant."

There are no commitments contained in this report.

Questions concerning this report may be addressed to Mr. Gordon J. Brownell at (315) 349-6360.

Very truly yours,

MICHAEL J. COLOMB

MJC:GJB:las
Enclosure

cc: USNRC, Region 1
USNRC, Project Directorate
USNRC Resident Inspector
INPO Records Center

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PDR ADOCK 05000333

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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TITLE (4)

Steam Leakage Detection System Outside Design Bases

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
12	06	99	99	013	00	01	05	00	N/A	05000
									N/A	05000

OPERATING MODE (9): N

POWER LEVEL (10): 100

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

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Mr. Gordon J. Brownell, Sr. Licensing Engineer	(315) 349-6360

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On December 06, 1999, with the mode switch in the "RUN" position and the plant operating at approximately 100 percent power, a partially completed Engineering calculation determined that certain steam leakage detection (SLD) systems were not capable of detecting steam leakage rates as specified in the Final Safety Analysis Report (FSAR). The FSAR states that the leakage detection systems are able to detect a 7 gallon per minute (gpm) steam leak for areas outside of the Primary Containment. Contrary to this, the results of the calculation show that a 7 gpm steam leak in the Reactor Water Cleanup (RWCU) System heat exchanger room, RWCU System "B" pump room, Residual Heat Removal System "A" heat exchanger room and the Main Steam tunnel would not be detected under most conditions.

The causes for the failure of the SLD system to meet design bases requirements were inadequate design verification and failure to implement original system performance requirements. A contributing cause was less than adequate basis for preparation of a written report.

Corrective actions include completing the current Engineering calculation, reviewing the need for performing additional area leakage detection calculations, and initiating a Nuclear Safety Evaluation (10 CFR 50.59) to change the plant design bases.

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EIIS Codes are in [ ]

EVENT DESCRIPTION

On December 06, 1999, with the mode switch in the "RUN" position and the plant operating at approximately 100 percent power, a partially completed Engineering calculation determined that certain steam leakage detection (SLD) systems [IJ] located outside of the Primary Containment [NH], which support the Primary Containment Isolation System [JM] (PCIS), were not capable of detecting steam leak rates as specified in the Final Safety Analysis Report (FSAR). The FSAR states that the leak detection capability for certain steam leakage detection systems is 7 gallons per minute (gpm) for areas including the Reactor Water Cleanup (RWCU) System [CE] heat exchanger [HX] room, RWCU System "B" pump [P] room, Residual Heat Removal (RHR) System [BO] "A" heat exchanger room, and the Main Steam tunnel.

The plant is designed with leakage detection systems which detect abnormal leakage from the Reactor Coolant pressure boundaries both inside and outside the Primary Containment. The systems are designed to ensure that conditions indicative of a failure of the Reactor pressure boundary are detected with sufficient timeliness and sensitivity to the extent feasible and practical.

The steam leakage detection systems outside the Primary Containment are comprised of ambient temperature sensors arranged for small leak detection (leakage rates less than the established leakage limits). Systems are designed such that high ambient temperatures initiate an alarm or isolation when temperatures reach a set point (less than or equal to 40 degrees Fahrenheit above ambient) which is indicative of a leak within the monitored area equal to the leakage rate criteria of 7 gpm.

Engineering concluded that the 7 gpm criteria was based on an early General Electric design guide/specification. However, the assumptions and accuracy of the design temperature setpoint values (area temperature changes resulting from a pipe crack) were based on a leakage detection system designed to use a differential thermocouple temperature sensing scheme (area ventilation entry and exit points). The 7 gpm criteria was therefore associated with a differential temperature scheme and not an ambient temperature scheme as used at FitzPatrick. This improper implementation of information from the design guide is a significant contributing factor as to why the 7 gpm requirement is not being met.

EVENT DESCRIPTION (cont'd.)

The Nuclear Steam Supply System (NSSS) vendor has established a 25 gpm leakage value for detecting and isolating a leak before a pipe crack propagates to a critical point. This value is used for SLD systems at other boiling water reactors (BWRs). Since this condition was identified, it has been determined that the SLD systems outside the Primary Containment, in most cases, are capable of detecting a 25 gpm leak. Therefore, this value can be applied to FitzPatrick's SLD systems. The exceptions are the Main Steam tunnel and RWCU System heat exchanger areas. The FSAR states that the Main Steam tunnel may have a leak as high as 150 gpm without reaching critical pipe crack length due to pipe size/diameter and associated pipe wall thickness. Therefore, Main Steam tunnel leakage detection values will be revised based on the FSAR value. The RWCU System heat exchanger area includes piping with minimum pipe size and pipe wall thickness larger than that assumed in the NSSS vendor's 25 gpm leakage case. Therefore, leakage detection values in this area will be above 25 gpm based on pipe size considerations. Consequently, the SLD systems outside the Primary Containment were determined to be operable.

CAUSE OF EVENT

The causes for the failure of the SLD system to meet design bases requirements were:

Inadequate Design Verification - A 1969 General Electric design guide was used in establishing SLD system design values, including the 7 gpm leakage detection criteria, used in the various locations outside the Primary Containment. The 7 gpm is also referenced in both the Final Safety Analysis Report (FSAR) and Technical Specifications (T.S.). It appears that the design guide was not applied properly to the original FitzPatrick plant design, and the plant designer failed to verify the SLD design guide assumptions with formal calculations for any of the individual SLD areas.

Apparent Lack of Implementation of an FSAR Statement - FSAR Section 4.10.3.4, "Leakage Detection System", subsection "Visual/Audible Inspection and Operability/Sensitivity Test" includes specific calibration references and test requirements to the SLD system's temperature sensors. It states in part, "With station in operation determine normal ventilation pattern and ambient temperature levels. Measure ventilation flow and calculate expected temperature rise for a 7 gpm leak from the system monitored. Set alarms or trip as required." Reviews were unable to find completed calculations to support temperature rise for a 7 gpm leak for any SLD system areas outside the Primary Containment.

CAUSE OF EVENT (cont'd.)

A contributing cause to this event was a less than adequate basis for preparation of a written report. T.S. Section 3.6, "Reactor Coolant System" BASES states in part "It is estimated that the Main Steam line tunnel leakage detectors are capable of detecting a leak on the order of 3,500 lb/hr. The system performance will be evaluated during the first five years of plant operation, and the conclusions of the evaluation will be reported to the NRC." The 3,500 pounds per hour (lb/hr) is equivalent to 7 gpm. On 03/14/83, a submittal (JPN-83-25) was made to the NRC by the Authority referencing this commitment and included a brief evaluation and performance summary of the Main Steam tunnel SLD system. Following the partial completion of the recent Engineering calculation and research into the bases for the original design leakage value of 7 gpm, it was concluded that the contents of the submittal letter lacked sufficient basis to meet the original intent of the commitment. The letter did not discuss any analysis or verification that the Main Steam tunnel SLD system could detect a leak on the order of 3,500 lb/hr.

ANALYSIS

This event is reportable under the provisions of 10 CFR 50.73(a)(2)(ii)(B), "Any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being in a condition that was outside the design basis of the plant."

FSAR Sections 4.10,"Reactor Coolant System Leakage Detection and Leakage Rate Limits" and 7.3,"Primary Containment and Reactor Vessel Isolation Control System", and T.S. Section 3.6,"Reactor Coolant System" BASES include requirements that relate to the SLD systems located outside the Primary Containment, being able to detect a 7 gpm (3,500 lb/hr) steam leak.

The purpose of the 7 gpm value was to assure early leak detection, and assure that a pipe's critical crack length is not exceeded. Ambient area thermal sensors are set to detect high temperature conditions with alarm and isolation limits being below the leakage rates where pipe leaks could potentially become pipe breaks. This alarm provides the plant sufficient time for corrective action before the Reactor Coolant pressure boundary could be significantly compromised.

A SLD system operability determination was completed for areas outside the Primary Containment following discovery of the design error. This analysis concluded that the SLD systems were operable based on their ability to detect and isolate leakage before reaching the critical crack length beyond which the crack is expected to propagate rapidly and the pipe is assumed to fail.

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Analysis (cont'd.)

The most relevant T.S. requirements (Table 3.2-1, Primary Containment Isolation System Instrumentation Requirements) indicate a trip level setting of less than or equal to 40 degrees Fahrenheit (F) above the maximum ambient area temperature for Main Steam tunnel, RWCU System, and RHR heat exchanger areas identified. The maximum ambient area temperature T.S. requirements are being met and are unaffected by the calculation. Engineering has concluded that steam leakage required to raise SLD system ambient area temperatures to an alarm/isolation level will be detected with sufficient timeliness to prevent a pipe failure. Therefore, the safety significance of this event is low because the SLD systems would have performed their alarm/isolation functions.

CORRECTIVE ACTIONS

1. Complete Engineering calculation (JAF-CALC-PC-03300) that is currently being performed for the steam leakage detection for the RWCU System pump room "B", the RWCU System heat exchanger room, the RHR System heat exchanger room "A", and the Main Steam tunnel. **(Scheduled Completion Date - 5/31/2000)**

2. Engineering will evaluate the need for performing additional calculations to other steam leakage detection areas outside the Primary Containment. **(Scheduled Completion Date - 5/31/2000)**

3. A 10 CFR 50.59 Nuclear Safety Evaluation will be completed to revise the JAF design bases for the SLD system from being able to detect a 7 gpm steam leak to being able to detect a 25 gpm steam leak. This will apply to RWCU System and RHR System areas outside the Primary Containment. The steam leakage detection value for the Main Steam tunnel area will be revised to 100 gpm. The Evaluation will include necessary changes to the FSAR and Technical Specifications Bases. **(Scheduled Completion Date - 5/31/2001)**

   Note: The completion of Corrective Actions 1, 2, and 3 along with flow and temperature measurements that have already been taken will satisfy FSAR Section 4.10.3.4, "Leakage Detection System", test and measurement requirements.

4. Any modifications to the facility required to meet the revised steam leakage detection criteria will be completed prior to start-up from the 15th Refueling Outage

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TEXT CONTINUATION

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CORRECTIVE ACTIONS (cont'd.)

5. The Authority will issue a revised submittal to the NRC to update previous letter (JPN-83-25 dated 03/14/83) and revise system performance evaluation results of the Main Steam tunnel leakage detection capabilities.
(Scheduled Completion Date - 6/15/2000)

ADDITIONAL INFORMATION

A. Previous Similar Events: NONE

B. Failed Equipment: NONE

C. Extent of Condition:

The conditions reported in this LER apply to all SLD system monitored areas outside the Primary Containment. The RWCU System pump room "B" is similar in volume, geometry and ventilation flow as RWCU System pump room "A", and the RHR System heat exchanger Room "A" is also similar in volume, geometry and ventilation flow as RHR System heat exchanger room "B". Hence, the unfinished calculation results can be directly applied.

D. Safety System Functional Failure:

This event did not result in a safety system functional failure in accordance with NEI 99-02, Revision D.