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JAFP-01-0182

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop O-P1-17
Washington, D.C. 20555

Subject: **Docket No. 50-333**
LICENSEE EVENT REPORT: LER-99-013-02 (DER-99-02838)

Steam Leakage Detection System Outside Design Bases

Dear Sir:

This report was 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."

This revised report is being submitted upon determination that changes to the JAF design/license basis for the Steam Leakage Detection System will require prior NRC approval in accordance with 10 CFR 50.90.

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,

T. A. Sullivan

TAS:GJB:las
Enclosure

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

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NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/200
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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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
									N/A	05000
12	06	99	99	013	02	08	17	01	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 Brownell, Sr. Licensing Engineer	(315) 349-6360

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

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

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION	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, the Main Steam Tunnel, Reactor Core Isolation Cooling (RCIC) System Enclosure, Torus Access Room, Crescent area above the RCIC Enclosure, and the High Pressure Coolant Injection (HPCI) System turbine/pump area 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 poor maintenance of the design and licensing basis for the SLD system.

Corrective actions include completing reviews of SLD System locations challenging the 7 gpm leakage limit, completing Engineering calculations for these locations, and the submittal of a 10 CFR 50.90 application to amend the current SLD System License bases.

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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.

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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. The areas not capable of detecting 25 gpm in all cases are the Main Steam tunnel, the High Pressure Coolant Injection (HPCI) System [BJ] pump area, the area above the Reactor Core Isolation Cooling (RCIC) System [BN] enclosure and RWCU System heat exchanger areas.

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.

Poor Maintenance of Design and Licensing Basis Information for the Steam Leak Detection System - A review of Design and Licensing Basis Information for the Steam Leak Detection system was conducted subsequent to Revision 0 of this LER. This review determined that the UFSAR description is inconsistent in some cases and lacks sufficient detail to firmly establish design basis system performance requirements.

An example of poor maintenance of the Licensing Basis 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.

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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 system 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.

Evaluations for the Reactor Coolant System Leakage Detection System identified eight locations that could not meet the FSAR's 7 gpm steam leakage detection limit for worst case conditions. Locations included the RWCU System pump room "B", RWCU System heat exchanger room, RHR System heat exchanger room "A", Main Steam Tunnel, RCIC Enclosure, Torus Access Room, Crescent area above the RCIC Enclosure, and the HPCI turbine/pump area.

Based on this finding, an additional evaluation was conducted to support operability of the leakage detection system conservatively assuming a complete absence of temperature monitoring. The evaluation concluded that, although this condition lowered the number of available steam leakage detection methods, the safety design bases for detection was satisfied because sufficient diverse detection methods remain available. Specific additional methods of leak detection are reactor building sump high pump-out rate, high steam line flow, and visual and audible inspection. All these remain as available methods of steam line leakage detection. Therefore, it was concluded that this event had minimal safety significance.

EXTENT OF CONDITION

Engineering calculations were initiated to analyze and evaluate the capability of the SLD System at various steam leakage flow rates at various locations and orientations in each of the identified areas. The results of these calculations will be used for the reconstitution of the design and License bases for the SLD System. These changes will be submitted to the NRC in a 10 CFR 50.90 License amendment application.

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CORRECTIVE ACTIONS

1. Evaluations have been completed for the Reactor Coolant System Leakage Detection locations challenging the 7 gpm steam leakage detection limit. A total of eight (8) locations were identified:

   - RWCU System pump room “B”,
   - RWCU System heat exchanger room,
   - Main Steam tunnel,
   - Torus Access Area,
   - HPCI System turbine/pump Area,
   - RCIC System Enclosure Area,
   - Crescent Area above the RCIC System Enclosure, and
   - RHR System heat exchanger room “A”.

2. Engineering calculations are being completed to evaluate the thermal-hydraulic capabilities of the Steam Leakage Detection System in the above eight listed locations. Calculations for five of eight locations have been completed, the remaining 3 calculations have been prepared and are in the review/approval process.
   (Scheduled Completion Date –12/15/2001)

3. The results of the completed calculations will be used to reconstitute the design and licensing basis for the Steam Leakage detection system. The revised design and licensing bases will be submitted to the NRC in a 10 CFR 50.90 License amendment application.
   (Scheduled Completion Date – 01/31/2002)

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

D. Safety System Functional Failure:

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