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January 21, 1992
JAFP-92-0061

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

SUBJECT: DOCKET NO. 50-333
LICENSEE EVENT REPORT: 91-033-00 - Potential Torus Pressure Instrument Errors

Dear Sir:

This report is submitted in accordance with 10 CFR 50.73(a)(2)(ii)(B) and (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. Liseno by direction

RADFORD J. CONVERSE

RJC:WVC:lar

Enclosure

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

Cert#
P 064 705 439

IE22 1/1

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 5

TITLE (4): Primary Containment Pressure Suppression Chamber Pressure Sensing Errors Following Potenital Accident Conditions Due to Inadequate Design Review

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)
1 2	1 9	9 1	9 1	0 3 3	0 0	0 1	2 1	9 2		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)	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: W. VERNE CHILDS, SENIOR LICENSING ENGINEER

TELEPHONE NUMBER: AREA CODE 315 349-6071

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

SUPPLEMENTAL REPORT EXPECTED (14)

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

EXPECTED SUBMISSION DATE (15): MONTH 0 3 DAY 2 5 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 in normal shutdown with the mode switch in the refuel position when it was determined on 12/19/91 that under Loss of Coolant Accident conditions water may accumulate in primary containment [NH] pressure suppression chamber (torus) pressure sensing lines causing significant instrument error. The potentially effected instruments are for accident monitoring [IP], input to certain Safety Parameter Display System (SPDS) displays, and for control of reactor building-torus vacuum breaker isolation valves. Additional analysis will be conducted to determine the magnitude and rate at which the collection of water causes the potential instrument error to be generated. All of the effected instruments will be moved to locations where future collection of water in sensing lines will not be possible. In addition, the physical configuration of other instrument sensing lines communicating to the drywell atmosphere will be evaluated to determine if similar potential problems exist.

NRC Form 366 (9-83)

INTERIM REPORT

EIIS Codes are in []

Description

On December 19, 1991 with the plant in a normal shutdown condition and the mode switch in the refuel position, it was determined that during Loss of Coolant Accident (LOCA) conditions several primary containment [NH] pressure suppression chamber (torus) instruments could have errors introduced as a result of the condensation of water in instrument sensing line.

The instruments affected by the possible condensation of water in the sensing line are:

- Torus wide-range pressure transmitter 27PT-101B1. This pressure transmitter feeds control room torus pressure recorder 27PR-101B1 (which is part of post accident monitoring [IP] system) and plant computer [ID] point EPIC-A-1286 which is part of the Safety Parameter Display System (SPDS).

- Torus narrow-range pressure transmitter 27PT-101B. This pressure transmitter feeds control room torus pressure indicator 27PI-101A (which is part of the post accident monitoring [I system] and plant computer [ID] point EPIC-A-1294 which is p   the SPDS.

- Torus wide-range pressure transmitter 27PT-101A. This pressure transmitter feeds control room torus pressure recorder 27PR-101A (which is part of the post accident monitoring [IP] system) and plant computer [ID] point EPIC-A-705 which is part of SPDS.

- Torus pressure switch 27PS-110B which senses differential pressure between the torus and reactor building [NG] (secondary containment). Pressure switch 27PS-110B provides a signal to open primary contianment [NH] isolation valve 27AOV-101B when torus pressure is less than reactor building pressure by no more than 0.5 psi. This allows vacuum breaker 27VB-7 to open and relieve the vacuum in the primary containment.

The instrument sensing line connected to the torus contains a vertical leg of approximately 20 feet which is not sloped back toward the torus so that any condensation of water vapor will not drain to the torus. During design basis Loss of Coolant Accident (LOCA) conditions the atmosphere in the torus air space will consist of non-condensable gases (primarily nitrogen) and water vapor with a relative humidity of 100 percent at a temperature of approximately 209°F. At the same time the instrument sensing lines, which are routed to an instrument rack in the same general area as emergency core cooling system pumps and piping, will be a temperature of 104°F or lower.

This differential temperature of more than 100°F results in conditions in which water vapor in the torus air space will migrate by the process of diffusion to the internal volume of the sensing line where the partial pressure of the water vapor is lower due to the sensing line being colder. When the water vapor reaches the relatively cool sensing line, it will be condensed. This process will continue until the conditions that cause the different partial pressure of water vapor in the two volumes of concern (the torus air space and the internal volume of the sensing lines) is eliminated. The collected water in the tubing will, after filling a small drip leg at the lowest point in the tubing, cause the sensed torus pressure to be higher than the actual pressure. As a result, the torus pressure information provided to operators on control room [NA] control panels and SPDS displays could result in improper (or improperly timed) decisions concerning accident mitigation and management. Similarly, SPDS displays in the Technical Support Center and Emergency Operation Facility (EOF) [NC] would contain the same erroneous information.

The collection of water in the sensing line for the pressure switch which controls the opening of the reactor building-torus vacuum breaker isolation valve would result in a condition in which the vacuum breaker isolation valve would not open until the torus vacuum is greater than the Technical Specification 3.7.A.4.a limit of 0.5 psi. The redundant pressure switch (which controls the isolation valve for the redundant vacuum breaker valve) also contains a vertical leg of approximately 10 feet. As a result, a similar potential problem exists for the redundant instrumentation resulting in conditions where the operation of both vacuum breakers may not take place within the Technical Specification limit of equal to or less than 0.5 psi.

An examination of plant operating records reveals a similar condition exists during normal plant operation for a portion of each year. During winter months the torus water temperature is normally approximately 75°F. At the same time, the temperature of the vertical portion of the sensing line is approximately 10° to 12°F cooler. The differences in the partial pressure of water vapor in the two volumes of concern is much less than that which exists during and following a LOCA, resulting in a much slower diffusion of water vapor from the torus air space to the instrument sensing line. It should also be noted that the small drip leg (collection volume) at the lowest point in the instrument sensing line is periodically checked for the presence of water. This check for the presence of water is done once each three months during time periods when primary containment integrity is not required. Past checks for water have not indicated that any water actually condenses in the sensing line and collects in the drip leg.

Cause

Cause of the event was inadequate design review during original plant construction. Instruments 27PT-101B1, 27PT-101A, and 27PT-101B were installed after initial operation of the plant as part of the modifications required by the Three Mile Island Unit 2 accident. These modifications relied on the original construction design review and as a result also did not receive an adequate design review.

The rate at which water could accumulate, and thus the magnitude of the error versus time, is not known. An analysis will be performed to conservatively estimate the water collection rate, and this LER will be updated following the analysis.

Analysis

The event is considered to be a reportable event under 10 CFR 50.73(a)(2)(ii)(B), a condition outside the design basis, and 10 CFR 50.73(a)(2)(v)(C) and (D), conditions that alone could have prevented fulfillment of the safety function of systems to control the release of radioactive material or to mitigate the consequences of an accident.

The errors introduced in the accident monitoring instrumentation and/or SPDS could result in errors in the decision process for operator actions for initiation (or termination) of containment venting. The potential error in indicated torus pressure could also have an effect on emergency core cooling system performance. The Residual Heat Removal (RHR)/Low Pressure Coolant Injection (LPCI) systems [BO] and low pressure core spray systems [BM] take suction on the torus. During portions of the time period following a LOCA the pressure in the torus is depended on to provide a portion of the Net Positive Suction Head (NPSH) for these pumps. Termination of containment venting, based on pressure indication that is higher than the actual pressure, could result in operation of RHR/LPCI and/or low pressure core spray pumps with inadequate NPSH. In addition, the errors introduced into the pressure (vacuum) sensing for control of reactor building-torus vacuum breakers could result in containment negative pressure (vacuum) in excess of the design values and Technical Specification limits.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

Corrective Action

1. An analysis will be performed to determine the rate at which water will collect in the torus pressure sensing lines. Due date February 12, 1992.

2. All of the instruments which could be affected by collection of water in the sensing lines will be moved to locations and elevations to preclude the collection of water in the sensing line prior to start-up following the 1992 Refueling Outage. Due date March 24, 1992.

3. Drywell instrumentation will be evaluated to verify that sensing line configurations do not allow the collection of condensed water vapor which introduces pressure measurement error.

Additional Information

Failed Components: None

Previous Similar Events: No other LERs events involving inadequate design review potentially causing significant instrument error have occurred been submitted at this facility.