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February 16, 1990
JAFP-90-0151

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

SUBJECT: DOCKET NO. 50-333
LICENSEE EVENT REPORT: 90-001-00 - Reactor Scram During Instrument Calibration

Dear Sir:

This Licensee Event Report is submitted in accordance with 10 CFR 50.73.

Questions concerning this report may be addressed to Mr. Hamilton Fish at (315) 349-6013.

Very truly yours,

R. Liseno by direction

WILLIAM FERNANDEZ

WF:HCF:lar

Enclosure

cc: USNRC, Region I
INPO Records Center
American Nuclear Insurers
NRC Resident Inspector

Cert No ~~P733060~~ P733061119

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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)	DOCKET NUMBER (2)	PAGE (3)
JAMES A. FITZPATRICK NUCLEAR POWER PLANT	0 5 0 0 0 3 3 3	1 OF 0 6

TITLE (4) Reactor Scram Due to False Low Reactor Water Level Signal During Calibration of Level Instrumentation

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 1	1 9	9 0	9 0	— 0 0 1 —	0 0	0 2	1 6	9 0		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 1 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)	[X] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 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
Hamilton C. Fish	AREA CODE 3 1 5 — 3 4 9 — 6 0 1 3

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)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE)	[X] NO	EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR

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

EIIS Codes are in []

A reactor scram from full power occurred at 10:45 A.M. on January 19, 1990 during the calibration of reactor water level instrumentation. The instrument which was being calibrated shares common reference and variable legs with instruments of the Reactor Protection System (RPS) [JE]. While isolating the instrument under test, a valve packing leak developed. During the response to this leak, valve actuation caused a false low water level to be sensed by the RPS. This false low level transient was caused by rapid valve movement by the field technician performing the valve manipulations. Corrective actions include review of this event with all I&C technicians, grooming of the communication system to eliminate noise, and repair of equipment that malfunctioned during the transient.

Related LERs: 85-012 and 87-020

EIIS Codes in []

Description

A spurious low reactor water level signal from the narrow range reactor pressure vessel level instrument channels 02-3LT-101A and B resulted in an automatic reactor scram from full power at 10:45 A.M. on January 19, 1990.

Immediately prior to the scram Instrument Surveillance Procedure ISP-3-4, "Reactor Water Level Instrument Calibration", was in progress. The purpose of the surveillance is to demonstrate semi-annually the operability and calibration of the reactor water level instrumentation in accordance with Technical Specification Table 4.2-6. The procedure requires that the instrumentation be removed from service by hydraulically isolating it from common instrument reference legs and variable legs. On January 19, three reactor water level transmitters had been scheduled for an instrument channel calibration. The calibration had been completed for one of the three instruments and isolation of the second instrument was started. In accordance with instructions, the field technician closed the low pressure side isolation valve for reactor feedwater level control system transmitter 06LT-52C. The next step was to open the equalizing valve. As the equalizing valve was being opened, a significant leak was observed to be originating from the packing gland of the low pressure isolation valve which had just been closed. The field technician informed the control room technician of the leak and discussed possible response actions.

The possibility of tightening the packing was considered. However, the control room technician recommended that this not be done because the design of the packing gland on this type of valve requires that it be full open and backseated before adjustment can be performed. Opening the valve would have returned the instrument to service. This was discussed as a possible course of action along with the option to continue to isolate the instrument from the system. After approximately two minutes passed in the assessment of the impact of the leakage on the system and determining possible courses of action, the control room test technician directed the field technician to continue with the isolation of the instrument. The field technician responded by manipulating the equalizing valve, most likely first in the incorrect direction and then more quickly in the correct direction.

The reactor scrammed as the technician was manipulating the equalizing valve. After the scram, the as-found check of the valves determined that the low pressure side isolation valve was still closed and that the equalizing valve appeared to be open by one and one-half turns. Rapid manipulation of the equalizing valve is suspected of generating the spurious level signal that resulted in the reactor scram.

Subsequent to the scram it was verified that the actual reactor water level had remained constant. The only parameters changing were the vessel level signals associated with the shared hydraulic headers. All other instruments showed a normal level indication. The scram signal was initiated for low reactor water level by two transmitters (02-3LT-101A and B) which share a common reference column and variable leg with the level transmitter which was being calibrated. The scram signal was recorded at 10:45:48 with transmitter 101A being tripped for 0.058 seconds and 101B tripped for 0.063 seconds. A reactor water level change of 24 inches in 0.063 seconds is a physical impossibility in this situation.

At the time of the scram the A standby gas treatment system was out of service for maintenance.

The operators followed the scram procedure and verified that the Group II isolation (reactor water level 177 inches above Top of Active Fuel (TAF) signal) actions were completed including initiation of standby gas treatment system B and isolation of the reactor water clean-up system and reactor building ventilation systems. The level decrease resulting from the scram pressure transient continued to 145 inches above TAF which is approximately 19 inches above the initiation points for the high pressure coolant injection [BJ], reactor core isolation cooling [BN], and trip of the reactor recirculation pumps [AD]. Accordingly, these systems did not activate and the recirculation pumps did not trip. The reactor recirculation pumps did automatically run back to the minimum flow condition as expected. The operator took manual control of the feedwater, manually tripping reactor feed pump (RFP) B when the level recovered to 190 inches above TAF. At 205 inches above TAF the operator took manual control of RFP A which subsequently tripped as designed at 225 inches above TAF. The RFP A and B discharge valves were closed by the operator. Reactor feed pump B was returned to service and the low flow control valve was placed in service to maintain vessel level.

The reactor water clean-up and reactor building ventilation systems were restored to service. A normal plant cooldown was initiated. After scram completion, while the operators were verifying that all rods were inserted, control rod [AA] 30-07 was found at position 02 instead of full in at position 00. It was then manually inserted.

At the time of the scram the computer display for drywell temperature was indicating high (178° F) because of instrument testing which was in progress. At the request of the shift supervisor, testing was suspended and the display was restored to a normal value of approximately 115° F.

Following the reactor scram and the associated trip of the turbine generator, the 10700 bus which is powered from the main generator tripped as expected. Among the equipment served by that bus is service water system [KG] pump C which also tripped as expected. Low service water pressure resulted in the automatic start of service water pump B. Twenty-five minutes after the plant trip at 11:10 A.M., a bus voltage transient resulted from an electrical short circuit and smoking of the motor for service water pump B. Due to the voltage transient the Uninterruptible Power Supply (UPS) system [EF] transferred to the direct current power drive. Other electrical events which were related to this bus voltage transient were a lock-up of the control positioner (scoop tube) for reactor recirculation pump motor generator set B, trip of the motor controller for valve 31MOV-153 main steam supply to the reboiler, and failure of a motor controller for a turbine building unit cooler fan.

In addition, the plant process computer system special transient data file was full at the time of the scram. Therefore, it failed to record data which would have assisted in confirming the valve manipulations by the field technician and other events occurring during the course of the transient.

The reactor mode switch was placed in start-up at 2:50 A.M. on January 22, 1990 and the generator placed on line at 10:55 P.M. the same day.

Cause

A false and spurious low reactor water level signal which lasted for 0.063 seconds initiated the reactor scram signal. To determine the root cause of the scram, a recreation of the testing conditions was simulated at the instrument rack during the plant shutdown. The valve packing leak conditions could not be duplicated precisely because the available reactor pressure had been reduced to 400 psig during cooldown instead of the normal operating pressure of 1,000 psig which existed before the scram. Thus, the rate of leakage and the nature of the pressure transients developed during the simulated transient varied somewhat from the actual conditions which existed during plant operation.

Nevertheless, the recreation demonstrated that the leakage from the low pressure side valve packing results in spurious level signals of the instruments that shared the common reference and variable legs. Past experience has shown the rate at which the equalizing valve was rotated in the open and close directions results in the false level low reactor water level signal which could initiate a scram. The combined effect of the two is believed to be the cause of this reactor scram.

Complicating the performance of this calibration was high noise levels from two sources. The first was the high ambient noise level at the instrument rack resulting from normal flow of steam and water. This noise cannot be easily reduced. The second source is the amplification noise on the communication circuit headsets.

Analysis

As an automatic scram, this event is reportable under the provisions of 10 CFR 50.73(a)(2)(iv) which requires reporting of any event or condition that resulted in a manual or automatic actuation of any Engineered Safety Feature. The chain of events is provided in the description section.

FSAR Section 14, "Safety Analyses", was reviewed with respect to this event. The plant was observed to respond as required to an indication of vessel low water level. The isolations and trips were appropriate based on instrument signals received. Following the plant trip minor damage to some non-safety-related electrical equipment occurred due to a bus voltage transient caused by an electrical short circuit in the B service water pump. The transient was evaluated and the electrical system responded as designed. Those problems had no detrimental effect on the course of the normal plant cooldown following the scram.

The standby gas treatment system fan B started and ran properly with fan A out of service. Vessel pressure and level control were maintained within acceptable ranges. The lock-up of the B recirculation MG set scoop tube was expected due to the known effect of a voltage transient on the associated relay. This MG set also ran back to the No. 2 limiter, also an event believed to have been caused by the bus voltage transient.

The one control rod which did not completely insert automatically was subsequently fully inserted manually. The rod had automatically inserted to notch position 02. Experience has demonstrated that small number of control rods in BWR plants will occasionally fully insert and then bounce back out to position 02. The reactivity represented by this single rod being at the first notch position is such that even if it had not been successfully inserted manually, sufficient shutdown margin would have been maintained and there would have been no safety consequences to the plant.

Corrective Action

1. An attempt was made to recreate the events which led to the trip signal. This was done while the reactor was in the shutdown mode at approximately 400 psig reactor pressure (compared to 1,000 psig when the event occurred). The recreation did recreate transients sufficient to have caused a scram.

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

2. The cause of the failure of the computer system to record all of the required data for post trip evaluation was determined and corrected.

3. The plant communication system is being groomed to reduce internal sources of noise.

4. This event will be reviewed as part of the technician training program.

Additional Information

Related LERs: 85-012 and 87-020