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United States Nuclear Regulatory Commission
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REFERENCE: DOCKET NO. 50-333
LICENSEE EVENT REPORT: 89-020-01
Reactor Scram
EHC System Malfunction

Dear Sir:

Enclosed is a supplement to the Licensee Event Report which was submitted in accordance with 10 CFR 50.73(a)(2)(iv) on December 5, 1989.

This supplement provides the results of factory testing of the EHC system control boards and improvements to the computer system.

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

Very truly yours,

WILLIAM FERNANDEZ

WF:HCF:lar

Enclosure

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

Cert No P395 598 075

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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 6

TITLE (4): Reactor Scram Initiated By High Average Power Range Monitor Neutron Flux from Pressure Transient from Turbine Control Valve Closure Due to Turbine Control Failure

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
11	05	89	89	020	01	01	29	90		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 100

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: Hamilton C. Fish

TELEPHONE NUMBER: AREA CODE 315 349-6013

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
X	JJ	SC	0084	N					

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)

Update Report - Previous Report Date 12/05/89 - Results of EHC Board Tests

EIIS Codes are in []

A reactor scram occurred from full power at 3:23 P.M. on November 5, 1989. An unidentified failure in an electronic control card of the Electro-Hydraulic Control (EHC) [JJ] system for the main turbine [TA] is believed to have opened the bypass valves and closed the intercept and control valves. This reduction in steam flow caused a pressure transient resulting in a reactor high flux scram signal from the Average Power Range Monitor (APRM) [IG]. The High Pressure Coolant Injection (HPCI) [BJ] system was inoperable prior to the scram. The automatic features of the plant responded normally to the scram except that one safety relief valve passed a small amount of steam at a pressure 5 percent below its design lifting pressure. The reactor core isolation cooling (RCIC) [BN] system was used to restore reactor water level. One control rod was not fully inserted, requiring manual insertion from position 02. Selected electronic control cards were replaced in the EHC system. The plant was restarted 11/10/89, and scrammed 11/12/89 (LER-89-023) for unrelated reasons. The plant was restarted 11/13/89 and run at 25 percent power to observe the EHC system. It was shutdown 11/20/89 for further work on the EHC system. Following testing and replacement of additional electronic circuit boards, the plant was restarted on 11/22/89.

The circuit boards removed from the EHC system have been sent to the vendor for analysis and possible root cause determination. Factory testing showed that all nine analog speed control boards met original equipment standards. No defects were found.

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Description

An automatic reactor scram from 100 percent power was initiated by high (120 percent trip) neutron flux on the Average Power Range Monitors (APRM) [IG] at 3:23 P.M. on November 5, 1989. Shift turnover was in progress, but did not contribute to this event. No testing or maintenance evolutions were in progress. A seven-day Limiting Condition for Operation (LCO) was in effect for inoperability of the High Pressure Coolant Injection (HPCI) [BJ] turbine speed control unit (LER-89-019).

Just prior to the scram, the operators experienced a strong rumble and vibration in the control room. The pressure transient caused by closure of the turbine control valves collapsed the voids in the reactor coolant resulting in a neutron flux spike and a low reactor water level of 133.9 inches above Top of Active Fuel (TAF).

The reactor level decrease resulted in the Reactor Core Isolation Cooling (RCIC) [BN] system automatic initiation to restore reactor water level. In addition to the initiation of RCIC, the low water level signal resulted in the trip of both reactor recirculation water pumps [AD], isolation of the reactor water cleanup [CE], and reactor building ventilation [VA] systems, and starting of both standby gas treatment [BH] systems. These automatic actions were expected in accordance with the plant designed actions for low reactor water level. In accordance with system design, RCIC shutdown automatically together with both turbine driven reactor feed pumps [SJ] on high reactor water level to prevent damage to the pump turbines. Feed pump "B" was restarted to maintain vessel level.

The reactor water cleanup and reactor building ventilation systems were restored to service. A normal plant cooldown was initiated. The reactor water recirculation pumps were not started because the Technical Specification limits for the reactor vessel differential temperature could not be met.

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 position 00. It was then manually inserted.

Peak reactor pressure reached 1082 psig during the control valve closure transient which is below the design setpoint for any of the safety relief valves [AD]. Safety relief valve (SRV) "F" has a design setpoint of 1140 psig (58 psi above the transient pressure). The increase in recorded SRV exhaust pipe temperature for this valve indicates that it passed a quantity of steam above that normally associated with pilot valve leakage. However, the absence of alarms from the acoustic monitor and exhaust pipe temperature rate of rise instruments indicate that the valve did not actually lift. This is confirmed by the exhaust pipe temperature which was below that associated with the lifting of an SRV.

The plant process computer [IQ] lacked sufficient capacity to record all of the scram data for which its program was designed. As a result, the one second of data (running from 0.8 seconds before the scram to 0.2 seconds after the scram) critical to the evaluation of the cause of the scram was not recorded. However, sufficient data was recorded to determine that partial closure of the turbine control valves resulted in a momentary opening of the turbine bypass valves and a pressure transient (spike) which in turn caused a neutron flux spike due to a reactor steam void collapse. This sequence of events also explains the rumble and vibration experienced by the control room operators just prior to the scram.

The precise failure mechanism of the turbine electro-hydraulic control (EHC) [JJ] system could not be identified. After discussion with the EHC system vendor, nine electronic circuit control boards which were believed to have the greatest potential for this type of EHC failure were replaced as a precautionary measure.

The reactor was then restarted on November 10 and held at less than 150 psig for testing of the HPCI turbine. On November 12 reactor pressure was increased for testing of the safety relief valves (SRV) [AD]. A scram occurred during the SRV testing (LER-89-023). After the post scram analysis and prestart-up testing, the reactor was brought critical on November 13 and connected to the grid on November 14. The plant was conservatively operated at 25 percent power to remain within the capacity of the turbine bypass valves to the main condenser to reduce the risk of a scram. Additional recording equipment was connected to monitor the performance of the EHC system at the reduced power. Some irregularities were observed and the vendor EHC expert was brought to the site on November 18. A normal plant shutdown was initiated on November 20 to permit performance of a more advanced test for inherent noise and to facilitate installation of upgraded electronic boards in the EHC system. The plant was restarted on November 22 with the generator connected to the grid on November 23. The plant was restored to operation at 100 percent power.

<u>Cause</u>

The scram originated from a high neutron flux signal from the 120% trip level of the Average Power Range Monitors. The cause of the high flux was the void collapse in the core moderator which resulted from a high steam pressure transient in the reactor vessel. This pressure transient was caused by closure of the turbine control and intercept valves and the inability to regulate reactor pressure because the steam flow was greater than 25 percent capacity of the bypass valves.

The closure of these valves is suspected to have resulted from an electronic noise generated turbine overspeed signal in the EHC system for the main turbine. This conjecture is based on the portions of control valve action which were recorded together with the experience of the EHC system vendor representative. The plant process computer transient data acquisition system became overloaded at the initiation of the event and failed to record some information during portions of the event. This limited the ability to reconstruct, follow, and analyze the transient positions of the control, bypass, and intercept valves, turbine speed, and EHC system performance.

This in turn limited the practical determination of the precise nature of the EHC failure. The independent EHC "first hit" panel indicated "Fast Closing Intercept Valves". These valves would normally close upon receipt of a turbine overspeed signal. The EHC system vendor noted that because the control valve and intercept valve position information was not available, it was extremely difficult to draw conclusions as to the actual cause of the turbine trip. Analysis of the data which was available shows that the turbine bypass valves opened by approximately 20 to 30 percent prior to the scram and that an additional pressure transient occurred after the scram signal and prior to the turbine trip. The data also shows that the electrical output of the main generator decreased prior to the turbine trip and prior to a decrease in total steam flow. There is no indication that the turbine control valve fast closure relays were actuated prior to the scram. This data is consistent with the opening of the bypass valves and supports a conclusion that the scram was caused by a failure in the EHC system. In addition, this data supports the rumble and vibration effects that were sensed by the operators prior to the scram.

During the subsequent five days of reduced power operation and testing, the additional recording instrumentation did show evidence of electronic "noise" spikes caused by external sources. Although an extensive testing program was implemented to locate the transient noise spike signal, no single circuit board could be identified as the source of the control system instability.

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 together with a description of malfunctioning equipment is provided in the description section. The HPCI system relays activated properly although the system was out of service at the time. The RCIC system activated automatically to maintain vessel water level. All systems activated in accordance with the assumptions of the Final Safety Analyses Report. The leakage of a small amount of steam by safety relief valve "F" at a pressure approximately five percent below the design lifting pressure did not represent an event significant to safety.

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 a 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

Short-Term:

Following the scram on November 5, the EHC load control and speed control boards were checked. No problems were found. Nine electronic control circuit boards which could have caused this type of failure were replaced. These included two each of the frequency to voltage converters and low value gates, four speed operation amplifiers, and one acceleration operational amplifier. All three speed pickups and coils were checked for grounds. The primary and backup speed pickup cables were subjected to a capacitance quality test and found to be satisfactory.

Following the five-day low power run, the plant was shutdown for further testing with the EHC vendor representative present. Updated models of the frequency to voltage electronic control boards were installed. Recorded noise signals were compared to those recorded prior to the five-day run. No differences were detected in the noise spectrum. The speed amplifier was adjusted to maintain the primary unit in control.

The operating topworks mechanism was replaced on safety relief valve "F" during the first shutdown period.

The computer programming was changed to increase the priority of the transient data recording program. Slight increases in total computer utilization capacity were obtained by increasing the time intervals between chemistry data point calibrations (from two minutes to six minutes) and between data transmission updates from once per minute to once every five minutes to other computer systems such as the NRC Emergency Response Data System (ERDS) and the plant data reporting system which serves remote terminals in the plant, emergency operations facility, technical support center, and the corporate office.

Long-Term:

1. Additional modifications to the EHC system are being pursued as recommended by the vendor expert to increase the overall reliability of the system.

2. The circuit boards which were removed from the EHC system were sent to the vendor for testing, analysis, and possible determination of the root cause of the failure. The testing has been completed. No defects were found. All boards met factory standards for the model year in which they were manufactured.

3. The computer system vendor is studying possible changes to the existing computer programs which will enhance the data gathering ability of the computer during scram events.

4. As a result of a subsequent scram on January 19, 1990 (LER-90-001) in which data was not recorded (coincident with the tape drive system being off line for service), a transient data file monitoring has been implemented. This system will alert the plant staff to a full file condition so that timely action can be taken to clear the file and assure continued availability to receive transient data.

Additional Information: Supplement 01 provides the results of factory testing of EHC control boards and describes improvements made to the computer system.