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RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 93-001-00:on 930128,phase-to-phase fault in MFW pump motor caused Bus 1 & 2 undervoltage condition resulting in automatic reactor trip.On 930129,another reactor trip occured.MFW Pump B motor replaced.W/930301 ltr.

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U. S. Nuclear Regulatory Commission
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Ladies/Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Reportable Occurrence 93-001-00

In accordance with the requirements of 10 CFR 50.73, "Licensee Event Report System," the attached Licensee Event Report for reportable occurrence 93-001-00 is being submitted.

Sincerely,

C.A. Schrock

C.A. Schrock
Manager-Nuclear Engineering

RPP/cjt

Attach.

cc - INPO Records Center
Mr. Patrick Castleman, US NRC
US NRC, Region III

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NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

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TITLE (4) Phase-to-Phase Fault in Main Feedwater Pump Motor Causes Bus 1 and 2 Undervoltage Condition Resulting in Automatic Reactor Trip

EVENT DATE (8)			LER NUMBER (8)				REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR		SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
										N/A	0 5 0 0 0
0 1	2 8	9 3	9 3	-	0 0 1	- 0 0	0 3	0 1	9 3		0 5 0 0 0

OPERATING MODE (9)	POWER LEVEL (10)	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
N	1 0 0	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
Dennis Rozell - Plant Nuclear Engineer	AREA CODE 4 1 1 4 3 8 8 - 2 5 6 0

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	S J	M O	A 1 8 0	Y							

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 fifteen single space typewritten lines) (16)

On January 28, 1993, at 0443 hours, with the plant operating at 100% power, a large fault on main feedwater pump B motor caused an undervoltage condition on Bus 1 and Bus 2. The bus voltage dropped below the undervoltage reactor trip setpoint and initiated an automatic reactor trip. The voltage on both buses quickly recovered once the fault was isolated.

On January 29, 1993, during the subsequent plant startup, another reactor trip occurred at 1516 hours from 5% power. The narrow range level for steam generator B reached the low-low level reactor trip setpoint of 17% while steam generator level was being manually controlled. Problems with steam dump valve SD-11A1 (to the condenser) contributed to the difficulty in controlling steam generator level.

During both trips, all plant systems responded as designed.

Corrective actions included replacing main feedwater pump B motor with a spare and using the steam generator power operated relief valves during subsequent plant startups until the steam dump valve is repaired. Steam dump valve SD-11A1 will be repaired during the 1993 refueling outage which is scheduled to start March 5.

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

Description of Event

On January 28, 1993, at 0443 hours, the plant was operating at 100% power when an automatic reactor [RCT] trip occurred due to low voltage on 4160V Buses 1 and 2 [BU]. Bus 1 and 2 supply power to the two main feedwater pumps (1 per bus) and the two reactor coolant pumps (1 per bus). A large fault on the main feedwater pump B motor [MO] caused an instantaneous overcurrent relay [50] to actuate which tripped the pump source breaker [BRK]. Due to the size of the fault, voltage on bus 1 and bus 2 dropped below the undervoltage reactor trip setpoint of 3200 volts and an automatic reactor trip was initiated. Voltage on bus 1 and bus 2 quickly returned to normal once the main feedwater pump B source breaker opened and cleared the fault. Main feedwater pump A and the reactor coolant pumps remained running due to a time delay in their source breaker trip circuitry.

The operators immediately entered Emergency Operating Procedure E-0, "Reactor Trip or Safety Injection" and, after verifying that safety injection was not required, transitioned at step 4 to ES-0.1, "Reactor Trip Response".

After the trip, the two motor-driven auxiliary feedwater pumps [P] and the turbine-driven auxiliary feedwater pump [P] started as designed, since the water level in both steam generators [SG] reached the low-low steam generator water level setpoint of 17%. Steam generator blowdown isolated upon the start of the auxiliary feedwater pumps as designed. Letdown

isolation occurred due to the reactor coolant system temperature falling below 540°F which caused pressurizer [PZR] level to decrease below the isolation setpoint. These events are considered automatic Engineered Safety Features (ESF) actuations. Charging flow was increased, pressurizer level recovered, and letdown flow was re-established approximately 15 minutes later. After verifying all systems responded as expected, a post trip review was performed to ensure that the event was analyzed and that the plant could be safely restarted.

On January 29, 1993, during the subsequent plant startup, another reactor trip occurred at 1516 hours when the unit was at 5% power. At the time of the plant trip, operators were performing actions in accordance with operating procedure N-O-02, "Plant Startup From Hot Shutdown to 15% Power". The following conditions existed at the time of the trip:

- The steam dump system (to the condenser) was being used to provide a steam load to increase reactor power above 5%,

- The steam dump system was in manual control. The operator had just switched to manual after observing erratic operation while in automatic during the power increase,

- Preparations were being made to latch the turbine,

- Main feedwater pump B was out of service for motor replacement,

- Feedwater flow to both steam generators (S/G) had recently been transferred from the auxiliary feedwater pumps to main feedwater pump A [P], and

- Steam generator levels were being manually controlled by the operator using the main feedwater bypass valves [FCV] in accordance with operating procedure N-FW-05A, "Feedwater System Normal Operation".

While controlling the S/G levels manually, the operator had reversed an increasing water level in both S/G's by throttling back on feedwater flow. The indicated narrow range level in S/G B had reached approximately 48% while the maximum indicated level in S/G A was approximately 14% higher at 62%. The operator was adjusting the steam dump valves [PCV] to reduce steam flow from the generators in order to stabilize S/G levels and steam flows. When the operator noticed that the levels in both generators were dropping faster than expected, due to a larger than expected shrink, he adjusted feedwater flow to compensate. The increased feedwater flow could not compensate for the shrink in the steam generators and the narrow range level for steam generator B reached 16%, which is below the low-low level reactor trip setpoint of 17%. The low-low level trip setpoint was not reached on S/G A due to the higher initial water level in the generator during this time.

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

The operators immediately entered Emergency Operating Procedure E-0 and, at step 4, transitioned into ES-0.1. The auxiliary feedwater system was still operating since the system had not yet been secured after the transfer to the main feedwater system.

After verifying that all plant systems responded as designed, a post trip review was performed to ensure that the event was analyzed and that the plant could be safely restarted. A reactor startup commeneed and the plant was connected to the grid at 2200 hours on January 29, 1993.

Cause of Event

The January 28 reactor trip was caused by a phase-to-phase fault on the main feedwater pump B motor stator resulting in a pump trip on instantaneous overcurrent. Due to the size of the fault, the voltage on 4160 volt Buses 1 and 2 dropped below the undervoltage reactor trip setpoint of 3200 volts. The reactor trip circuitry relays completed their logic sequence and initiated a reactor trip before the voltage on Buses 1 and 2 could recover after the fault at the pump B motor was cleared. The bus 1 and 2 undervoltage relays [27] protect the reactor coolant pumps [P] from undervoltage conditions by tripping open the pump source breaker [BKR]. The bus 1 and 2 undervoltage reactor trip shuts down the reactor in anticipation of losing one or both reactor coolant pumps and therefore reactor coolant flow. The reactor coolant pumps were not tripped because there is a time delay in the breaker trip circuitry which allows for brief (0.1 second or less) voltage drops on bus 1 and bus 2.

A root cause analysis was performed in an attempt to determine the exact cause of the fault, but a specific eause could not be determined due to the damage which resulted from the fault. There were also no indications of a degrading condition on the motor. Each year, during the refueling outage, testing is performed on the main feedwater pump motors to determine any degrading trends which would affect pump perfornance. This testing of the motor windings measures insulation degradation and resistance. The 1992 results showed no degrading trends in the motor. The motor was installed approximately 4.5 years ago and was new at that time. It was scheduled to be replaced with the spare motor this outage (1993) and overhauled (inspected and tested by a vendor) as part of its 5 year preventive maintenance schedule.

The cause of the January 29 reactor trip is a combination of a problem with a steam dump valve and the effect this had on manual control of S/G level. Prior to the trip, the Instrumentation and Control (I&C) Field Supervisor went to check the steam dump valves locally because of the stated erratic operation of the steam dump system in automatic. Upon inspecting the steam dump valves he notieed that one of the steam dump valves, SD-11A1, appeared to be pulsating during operation. As the I&C Field Supervisor returned to the control room with this information, the rcactor trip occurred.

The steam dump valve oscillations contributed to de-stabilizing the plant while manually recovering steam generator level. Manually controlling S/G level is a challenging evolution for an operator under normal situations. The shrink and swell phenomena in a U-tube steam generator can drastically affect S/G narrow range level indications with small steam or feedwater flow adjustments. The abnormal operation of SD-11A1 caused excessive steam to be dumped from the steam generators which caused S/G levels to shrink more than the operator expected. The narrow range water level of S/G B at this point was low enough that the shrinkage caused the low-low level trip setpoint to be reached.

Analysis of Event

For the January 28 reactor trip, a review of the Sequential Event Recorder [CPU] (SER) printout for the time period prior to the reactor trip shows that the "Feedwater Pump B Breaker Overcurrent Trip" SER alarm point [ANN] was the first to actuate. This was followed by the "Bus 1 or 2 Voltage Low Alarm" alert SER point and the "Feedwater Pump B BKR 1-202 Open" SER alarm point. The "Bus 1 and 2 Undervoltage Reactor Trip" SER alarm was received immediately after the pump breaker open alarm was received. To actuate the bus 1 or 2 low voltage alert SER point, one of the two undervoltage channels on bus 1 or bus 2 must actuate. The reactor trip is initiated when one of the two channels on both buses see an undervoltage condition. Once the feedwater pump B motor breaker opened and cleared the fault, the voltage recovered on bus 1 and bus 2 prior to tripping the reactor coolant pump breaker.

This allowed the reactor coolant pumps and main feedwater pump A to continue to operate. Thirty seconds after the reactor trip, the turbine generator time delay trip actuated the lockout relays and initiated bus 1 through 4 automatic voltage restoring logic. The automatic voltage restoring logic performed as designed and transferred power for buses 1 through 4 from the main to the reserve auxiliary transformer.

The detailed review of the SER printout verified that the feedwater pump protection circuitry and the reactor trip circuitry responded correctly. This review considered the potential size of the fault and the range of actuation times of the various relays in each circuit. All systems in the plant responded as designed and, therefore, there were no safety consequences resulting from this event.

During the startup on January 29, the steam dump system was being used to provide a steam load on the plant. Using the steam dump system instead of the S/G Power Operated Relief Valves [RV] (PORVs) conserves the amount of makeup water that would be required during startup and also provides a more predictable plant response due to using an equal steam load from each steam generator. The operator encountered difficulty in controlling S/G level during this startup because of the pulsation problems with steam dump valve SD-11A1. This resulted in abnormal steam flow changes in response to operator demands and the resulting low-low S/G level reactor trip.

With the exception of the erratic operation of the steam dump valve, all other plant systems functioned as designed and, therefore, there were no safety consequences resulting from this event.

Both reactor trips are reportable in accordance with 10CFR50.73(a)(2)(iv) as an event that resulted in actuation of the reactor protection system. These events were also reported in accordance with 10CFR50.72(b)(2)(ii) at 0616 hours on January 28, 1993, for the initial trip and 1627 hours on January 29, 1993 for the trip during plant startup. The ESF actuations which occurred after the January 28 reactor trip were reported in accordance with 10CFR50.72(b)(2)(ii) at 0738 on the same day. These ESF actuations are also reportable in accordance with 10CFR50.73(a)(2)(iv) as an event that resulted in an ESF actuation.

Corrective Actions

January 28, 1993 trip:

1. A post trip review was conducted to verify that the plant responded as designed.

2. The motor for main feedwater pump B was replaced with a spare.

3. An engineering support request was initiated to investigate increasing the undervoltage time delay setpoints for bus 1 and 2. This may prevent causing another reactor trip due to a large fault on bus 1 or 2.

4. The failed motor will be rebuilt and used as a spare. It was originally scheduled to be replaced with the spare during the 1993 refueling outage and then overhauled (tested, inspected and repaired as needed).

January 29, 1993 trip:

1. A vendor representative has been contacted and is scheduled to investigate the problem with the steam dump valve during the 1993 refueling outage. Repairs will be made based on the results of the investigation.

2. During the subsequent startup, the S/G PORVs were used to provide a steam load on the plant.

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Additional Information

EQUIPMENT FAILURES:

1. The motor for the B main feedwater pump is an Allis-Chalmers, Wound-Rotor Polyphase Motor, 3 Phase, 4000 Volt, Model No.05-687-084-530

SIMILAR EVENTS:

None