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SUBJECT: LER 89-010-00:on 890620,unplanned partial Group II isolation.
W/8 ltr.

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July 20, 1989

Report Required by:
10 CFR Part 50,
Section 50.73

U S Nuclear Regulatory Commission
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MONTICELLO NUCLEAR GENERATING PLANT
Docket Nos. 50-263 License Nos. DPR-22

Unplanned Partial Group II Isolation

The Licensee Event Report for this occurrence is attached.

This event was reported via the Emergency Notification System in accordance with 10 CFR Part 72, on June 20, 1989.

Thomas M Parker
Manager
Nuclear Support Services

c: Regional Administrator-III, NRC
NRR Project Manager, NRC
Resident Inspector, NRC
MPCA
Attn: J W Ferman

Attachment

IE22

NRC Form 366
(9-83)

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

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)
Unplanned Partial Group II Isolation

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 6	2 0	8 9	8 9	— 0 1 0 —	0 0	0 7	2 0	8 9		0 5 0 0 0
										0 5 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check one or more of the following)* (11)

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 0

	20.402(b)		20.405(c)	XX	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
Eric C. Sopkin, Site Superintendent	AREA CODE 6 1 2 / 2 9 5 - 1 0 5 1

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	J M	R L Y	G 0 8 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES *(If yes, complete EXPECTED SUBMISSION DATE)*	XX NO				

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

During cold shutdown conditions, a Residual Heat Removal (RHR) Low Pressure Coolant Injection (LPCI) Inboard Injection Valve closed in response to the de-energization of Primary Containment Isolation System (PCIS) Group II relay 16A-K17. This was an inadvertent isolation initiated by the removal of control power fuse 16A-F21 (FU) that resulted in the interruption of RHR Shutdown Cooling flow. The fuse was removed to verify that bypasses installed on PCIS Group II relay 16A-K21, which had experienced a random failure, did not affect operability of the associated PCIS Group II isolation logic. When the valve closed, the RHR Minimum Flow valve opened causing a decrease in reactor water level. The RHR pump was removed from service to close the RHR minimum Flow valve, preventing any further decrease in reactor water level. RHR Shutdown Cooling was subsequently restarted. Misinterpretation of a PCIS elementary drawing by the senior-licensed shift supervisor caused the event. Supplemental operator training and revision of the PCIS elementary drawing are planned to prevent recurrence.

DESCRIPTION

Initial plant conditions: Reactor was in cold shutdown with reactor water level +35 inches and reactor water temperature 167 degrees F. The #11 Residual Heat Removal (RHR) system was in service for Shutdown Cooling with flow to the #11 Reactor Recirculation system (RR) loop. Both Reactor Recirculation pumps (P) were operating at minimum speed. The Reactor Water Cleanup system (RWCU) was in service discharging to the main Condenser (COND). The #11 Control Rod Drive (CRD) pump (P) was in service supplying 40 gallons per minute (gpm) cooling water flow to the control rod drive mechanisms. The Primary Containment was inerted. Reactor level was being controlled by maintaining RWCU discharge flow balanced with CRD flow.

At 2305 on June 19, 1989, alarm C04-B-3, DRYWELL SUMP VALVES CLOSED, was received in the Control Room. Verification of the alarm condition by the Operating Crew revealed that all Drywell Equipment and Floor Drain Sump valves (ISV) indicated open on Control Panel C04. At 2315, a local observation of these valves confirmed that they were all open. At 2330 manual operation of the Drywell Equipment and Floor Drain Sump pumps (P) was attempted. It was determined that only the Drywell Floor Drain Sump pumps would operate.

The Operating Crew consulted Monticello Plant Technical Specifications and determined that this was not a Technical Specification related event. A review of the Primary Containment Isolation System (PCIS) elementary drawing (NSP Monticello 7823-4-16) indicated that a malfunction of control relay 16A-K21 (RLY) had occurred. At 2345 local inspection of Relay Cabinet C41 showed that the normally energized relay had failed electrically in the de-energized position. A burning odor was also noted in the vicinity of the relay cabinet. Continuity was verified across the associated control power fuse 16A-F21 (FU).

The function of control relay 16A-K21 is to stop and prevent operation of the Drywell Equipment Drain Sump pumps upon closure of the Drywell Equipment Drain Sump Pump Inboard Discharge valve (ISV) CV-2561A. This prevents pump operation against a shutoff head. Although CV-2561A was actually open, failure of control relay 16A-K21 would not permit operation of the Drywell Equipment Drain Sump pumps P20A and P20B. As a result, Drywell Equipment Drain Sump water level would continue to increase until it overflowed into the adjacent Drywell Floor Drain Sump. Additionally, the Drywell identified leakage detection instrumentation would not be available. This instrumentation utilizes Drywell Equipment Drain Sump pump running times as reference for initiation of Drywell Identified Leakage alarms. The instrumentation and alarms were not required by Technical Specifications under the existing plant conditions.

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 5/31/85

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

The Operating Crew used the PCIS elementary drawing (NSP Monticello 7823-4-16) as a reference for the preparation of a Jumper/Bypass Form for contact jumpering of control relay 16A-K21. The contact jumpering would permit operation of the Drywell Equipment Drain Sump pumps P20A and P20B pending replacement of the failed 16A-K21 control relay.

The Operating Crew also prepared a Work Request Authorization (WRA 89-50029) and an Isolation Order for replacement of control relay 16A-K21. The same PCIS elementary drawing (NSP Monticello 7823-4-16) was used for determination of the electrical isolation necessary to permit the control relay replacement. The Operating Crew determined that control power fuse 16A-F21 if removed would provide the electrical isolation required for WRA 89-50029. It was acknowledged that during the period of time that control power fuse 16A-F21 was removed, a partial PCIS Group I isolation would be enforced. From the PCIS elementary drawing (NSP 7823-4-16) this was determined to include the inboard RHR to radwaste valve MO-2032 (ISV) (which was currently closed and de-energized), Traversing In-core Probe (TIP) Ball valves TIP-1-1, TIP-2-1 and TIP-3-1 (ISV) (which were currently closed), and inboard Drywell Equipment and Floor Drain Sump Pump Discharge valves (ISV) CV-2541A and CV-2561A (which were currently open).

At 0020 on June 20, 1989, two members of the Operating Crew left the Control Room to install the jumpers on control relay 16A-K21 and temporarily remove and then reinstall control power fuse 16A-F21 to verify receipt of the partial PCIS Group II isolation. The second individual provided independent verification of these actions.

At 0026 the Control Room was notified that the jumpers had been installed on control relay 16A-K21 and control power fuse 16A-F21 had been removed and immediately reinstalled. The Operator at Control Room Panel C04 verified receipt of the partial PCIS Group II isolation by noting the previously identified affected valve positions on Control Room Panels C03, C04 and C36. The partial PCIS Group II isolation was then reset and the Drywell Equipment Drain Sump Inboard Pump Discharge valves CV-2541A and CV-2561A opened.
At 0027 alarm C03-A-10; RHR HX TUBE/SHELL LO DIF PRESS was received. The Operator responding to the alarm observed decreasing flow in the #11 RHR system and then the opening of the #11 RHR Minimum Flow valve (XCV) CV-1994. Another Operator located at Control Room Panel C05 reported that reactor water level was slowly decreasing. That Operator then reduced the setpoint on the RWCU Drain Flow Regulator for RWCU Dump Flow Control valve (XCV) CV-2403 to zero. The Operator located at Relay Cabinet C41 was instructed to remove the previously installed jumpers on control relay 16A-K21. These jumpers were removed immediately.

Due to a burned-out light bulb for the RHR Low Pressure Coolant Injection (LPCI) Inboard Injection valve (INV) MO-2014 closed indication, operation of this valve was not recognized until it reached the full closed position. At this point all valve indication for MO-2014 was extinguished. The Operator attempted to open MO-2014 two times. It would subsequently close automatically after reaching the full open position. The stroke time for MO-2014 is 60 seconds. When RHR flow would decrease below 600 gpm the #11 RHR pump (P) was shutdown and #11 RHR Minimum Flow valve CV-1994 closed per design.

During the previous review of the PCIS elementary drawing (NSP Monticello 7823-4-16) the RHR LPCI Inboard Injection valve MO-2014 was not identified to be associated with the logic string powered by control power fuse 16A-F21. Although the partial PCIS Group II isolation had been reset, MO-2014 Shutdown Cooling Group II Isolation Reset pushbutton had not been depressed. Depressing this pushbutton would have prevented the closure of RHR LPCI Inboard Injection valve MO-2014 after it was re-opened.

At the time of #11 RHR pump shutdown (0033), the Control Room Panel C05 Operator reported that reactor water level had stabilized at +25 inches and reactor water temperature was 167 degrees F. The reactor water heatup rate was estimated to be less than 0.5 degrees F per minute.

The RHR system was returned to Shutdown Cooling service within 16 minutes. Reactor water temperature was 173 degrees F upon restoration of RHR Shutdown Cooling flow. Reactor water level was restored to +35 inches coincidentally with resumption of RHR Shutdown Cooling. The RWCU system drain flow regulator was returned to service for reactor level control.

At 0350, a 10CFR50.72 report to the Nuclear Regulatory Commission (NRC) was made on the basis of a perceived PCIS Group II logic problem causing the RHR LPCI Inboard Injection valve MO-2014 to close.

At 0430 a review of the PCIS elementary drawing (NSP Monticello 7823-4-16) and RHR elementary drawings (NSP Monticello 7905-46-4) was completed by the Shift Technical Advisor. It was determined that the RHR LPCI Inboard Injection valve MO-2014 had operated per design in response to the partial PCIS Group II isolation.

At 0615 control relay 16A-K21 replacement was completed and post-maintenance testing of this control relay was performed successfully.

AT 0730 an additional review of the RHR LPCI Inboard Injection valve MO-2014 closure by Plant Technical Staff confirmed that the valve operated per design.

At 1030 a followup report was made to the NRC to clarify the earlier 10CFR50.72 notification.

CAUSE

The root cause of this event is licensed operator error, specifically the misinterpretation of a PCIS elementary drawing. The initial review of this drawing by the Operating Crew failed to identify the cause and effect relationship between the removal of control power fuse 16A-F21 and RHR LPCI Inboard Injection valve MO-2014. This was not a cognitive error.

The Operating Crew also failed to realize that testing of this nature must be performed under the provision of Monticello Plant Administrative control Directive 4 ACD-04.04, SPECIAL PROCEDURES.

A lack of clarity of the PCIS elementary drawing contributed to the error. The drawing does not identify the RHR LPCI Inboard Injection valve MO-2014 isolation directly, but specifies an "RHR TRIP INTERLOCK". During their review the Operating Crew interpreted the term RHR TRIP INTERLOCK as the isolation associated with the RHR To Radwaste Isolation valve MO-2032. The RHR to Radwaste Isolation valve MO-2032 isolation logic is displayed on the PCIS elementary drawing. The Operating Crew also failed to see the attendant reference associated with the RHR TRIP INTERLOCK.

There were no unusual characteristics of the work location that directly contributed to the error.

ANALYSIS

Although the intentions of the Operating Crew were conservative, the lack of a more thorough review of their actions led to a temporary interruption of Shutdown Cooling flow to the reactor. Due to the timeliness of their response actions reactor water level and temperature changes were minimized.

During the event reactor water level decreased from +35 inches to +25 inches. No other safety system initiated, other than the partial PCIS Group II isolation, as a result of this event. The reactor water temperature increased from 167 degrees F to 173 degrees F which did not pose a threat to exiting the cold shutdown condition. Both reactor water level and temperature were returned to their pre-event values within 16 minutes after termination of RHR Shutdown Cooling flow. RHR Shutdown Cooling capability remained available at all times since the redundant train of RHR Shutdown Cooling was available at all times.

As a result of this event no systems, trains, or components were rendered inoperable. At no time was there a threat to the public safety or departure from a previously analyzed plant condition.

Had the Operating Crew failed to take corrective actions in the response to this event, automatic system functions would have acted to terminate the reactor water level decrease at +9 inches. At that time a PCIS Group II Shutdown Cooling isolation, independent of the RHR LPCI Inboard Injection valve MO-2014 isolation, would have occurred. This isolation would result in the automatic closure of RHR Shutdown Cooling Inboard and Outboard Suction valves (ISV) MO-2029 and MO-2030, and the automatic tripping of the associated RHR Minimum Flow valve CV-1994 along with tripping of the RHR pump. Relay 16A-K21 does not provide an ESF function. Its function is to protect the Drywell Equipment and Floor Drain pumps from operating against a shutoff head. The relay fails safe and the subsequent effect was to prevent operation of the Drywell Equipment Drain Sump pumps.

CORRECTIVE ACTIONS

Following the event, the Operating Crew returned RHR Shutdown Cooling to normal service and restored reactor water level and temperature to pre-event values. The failed light bulb for RHR LPCI Inboard Injection valve MO-2014 was replaced during the event.

Followup actions included replacement of control relay 16A-K21. The replacement was completed and the plant returned to normal conditions by 0615 on June 20, 1989.

Other corrective actions include:

1. Supplemental training for Operating Crews to address their misinterpretation of the PCIS elementary drawing was completed on July 14, 1989.

2. A review of the PCIS elementary drawing to determine if revision is necessary to address the Operating Crews perceived lack of clarity will be completed by August 31, 1989.

3. Licensed operator print reading training will be revised to emphasize a symptom-based problem-solving approach and will be presented to all Operating Crew by November 16, 1989.

4. Additional licensed operator training in the administrative control of testing and bypasses will be developed and presented to all Operating Crew's by November 16, 1989.

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

ADDITIONAL INFORMATION

Failed Component Identification

The failed component was a General Electric Company type CR120A0400AA control relay, designation 16A-K21.

PREVIOUS SIMILAR EVENTS

None