NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

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TITLE (4)
Inadvertent Injection Of Water Into Reactor Vessel During Surveillance Test

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	0 9	8 8	8 8	— 0 1 8	— 0 0	0 7	1 1	8 8		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)	[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 (AREA CODE)
Ralph W. Krause	4 0 2 8 2 5 - 3 8 1 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

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)

On June 9, 1988, during the performance of Surveillance Procedure 6.3.4.3 (Sequential Loading of Emergency Diesel Generators) on Emergency Diesel Generator (DG)-2, approximately 15,000 gallons of Suppression Pool water were injected into the Reactor Vessel. The injection occurred when Residual Heat Removal (RHR) Pump D and Core Spray (CS) Pump B automatically started per the surveillance procedure, and their injection valves inadvertently opened because the Operator performing the test setup failed to secure power to the injection valves. The plant was in cold shutdown with the Reactor Vessel Head installed, and a refueling and maintenance outage in progress.

The principle causes for the event were attributed to the following three items. The procedure was inadequate in that the steps which directed the Operator to check the valve position and secure power had only one sign-off per valve (one sign-off for two actions). The Operations Crew was distracted during the test setup due to several observers in the Control Room and normal outage activities. Thirdly, the Operator failed to follow the procedure in that the valve position was checked, but the power not secured.

Immediate corrective actions were to terminate the injection and to revise the surveillance procedure to provide separate sign-offs to check the valve position and secure power. Additional corrective actions to prevent recurrence include reviewing the adequacy of Control Room conduct and manning procedures to limit the potential for distraction of Control Room Operators.

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

A. Event Description

On June 9, 1988, at approximately 2:10 P.M., during performance of Surveillance Procedure 6.3.4.3 (Sequential Loading of Emergency Diesel Generators) on Emergency Diesel Generator (DG)-2, approximately 15,000 gallons of Suppression Pool water were injected into the Reactor Vessel via Residual Heat Removal (RHR) Pump D and Core Spray (CS) Pump B. The plant was in Cold Shutdown condition with the Reactor Vessel Head on, and a refueling and maintenance outage in progress.

Surveillance Procedure 6.3.4.3 is performed once per cycle to functionally test the emergency start of each Emergency Diesel Generator, RHR pump and CS pump, as well as the loading sequence of safety-related equipment on the associated DG. A portion of the test setup is to close and de-energize the Reactor Vessel injection valves for RHR and CS, to prevent opening when the emergency start signal is simulated. The test also sets up several other RHR and CS valves, to prevent closure during the simulated start signal and allow a full flow test path to the Suppression Pool for the pumps during the test.

The test is initiated by simulating undervoltage on the 4160 volt critical bus, then immediately simulating low Reactor Vessel water level ($\geq$ -145.5"). The associated DG will start and come to rated speed and voltage while load shedding is automatically accomplished on the critical bus. Approximately ten seconds after the DG starts, if all breaker permissives are satisfied, the DG output circuit breaker will close to re-energize the critical bus, and one RHR pump will start. After a five second time delay, the second RHR pump powered from that bus will start. Approximately five seconds later, the CS pump powered from that bus will start. After the Licensed Operator verifies the pump starts, he is directed by procedure to open the test return valves to the Suppression Pool to establish rated flow. At five second intervals after the CS pump starts, a Service Water pump and Reactor Equipment Cooling pump will also start. It should be noted that the RHR pumps powered from a given DG are in opposite RHR loops; i.e., DG-1 powers RHR Pump A in Loop A and RHR Pump B in Loop B, and DG-2 powers RHR Pump C in Loop A and RHR Pump D in Loop B.

When the surveillance test was performed on June 9, 1988, the above sequence occurred with the following deviations. Immediately upon simulating the low Reactor Vessel water level, DC powered valve RHR-MOV-M025B (RHR Loop B Inboard Injection Valve) began to open. When the critical bus was re-energized by DG-2, valve RHR-MOV-M027B (RHR Loop B Outboard Injection Valve) began to open, and valve RHR-MOV-M039B (RHR Loop B Outboard to Suppression Pool Valve) began to close. These valve operations, along with RHR Pump D starting when the bus was re-energized, immediately created an injection path into the Reactor Vessel via RHR Loop B. Also, when the critical bus was re-energized, valves CS-MOV-M011B (CS Loop B Outboard Injection Valve) and CS-MOV-M012B

(CS Loop B Inboard Injection Valve) began to open. When CS Pump B started ten seconds later, an additional flow path to the Reactor Vessel through CS Loop B was established.

When the Licensed Operator monitoring RHR Loop B realized this loop was injecting, he attempted to reopen RHR-MOV-M039B to divert flow to the Suppression Pool and reclose RHR-MOV-M025B. Valve RHR-MOV-M025B received an open signal and valve RHR-MOV-M039B received a close signal when the low water level was simulated, and this signal remains in affect (sealed-in) as long as the initiation signal is present. After unsuccessfully attempting to reposition the valves, the Licensed Operator secured RHR Pump D to terminate injection. The Senior Licensed Operator monitoring CS Loop B, upon recognizing this loop was injecting, reduced the injection flow rate by opening valve CS-MOV-M026B (CS Loop B Test Return to Suppression Pool). The Senior Licensed Operator subsequently closed valve CS-MOV-M012B to terminate injection. The total injection time for RHR Loop B was 1 minute 37 seconds, and 53 seconds for CS. It is estimated RHR injected 11,000 to 12,000 gallons, while CS injected 3,000 to 4,000 gallons.

B. Plant Status

The plant had been shutdown for refueling and maintenance since March 5, 1988. At the time of the event, Surveillance Procedure 6.3.4.3 (Sequential Loading of Emergency Diesel Generators) was in progress on Emergency Diesel Generator - 2.

C. Basis for Report

Unplanned actuation of an Engineered Safety Feature (ESF) (water injection into Reactor Vessel), reportable in accordance with 10CFR50.73(a)(2)(iv).

D. Cause

Procedural inadequacy. A comparison was made of the revision used on June 9 (an upgraded version, first time used) to the previous revision (used several times in the past). While the comparison failed to disclose any significant differences in this portion of the procedure, it was discovered that both versions allowed more than one action to occur per set of Operator initials.

Lack of mental attention. From discussions with the Control Room Operators, there were too many distractions during the conduct of the surveillance procedure. Several observers (INPO and NRC) were present in the Control Room and were asking numerous questions regarding the test. In addition, the surveillance test was being performed on the day shift

during an outage, where other evolutions in progress added to the distractions. The cumulative result was a lack of mental attention on the part of the Operators.

Procedure not followed. Surveillance Procedure 6.3.4.3, Table 2, entitled "DG-2 Valve Line-up and Power Status" required the Operator to check the valve position of RHR and CS injection valves closed and the RHR Suppression Pool outboard valves open, and ensure the power supply to each of the valve operators is secured. The Operator verified the valve position, but failed to verify that power was secured.

E. Safety Consequences

None. The Emergency Diesel Generator, Core Spray pump, and Residual Heat Removal pumps operated as required per the procedure. The RHR and CS valves responded properly to the simulated injection signal.

F. Safety Implications

Had the injection continued with no Operator action, the Reactor Vessel would have completely filled and pressure would have stabilized at approximately 340 psig (shutoff head for Core Spray). This pressure is within the Safe Operating Region of Technical Specifications Figure 3.6.2, Minimum Temperature for Pressure Tests Such as Required by Section XI, for temperatures > 110°F. Actual Reactor Vessel metal temperatures ranged from 170°F to 190°F in the flange area, with no vessel temperatures below 125°F.

Surveillance Procedure 6.3.4.3 is typically performed within one to two weeks prior to plant startup from refueling outages. At this point, Reactor Vessel assembly is complete. Had this test been performed when the Reactor Vessel Head was off, and the reactor flooded for refueling, the reactor cavity, spent fuel storage pool, and steam dryer/moisture separator pool would not have overflowed. Nominal water level is 12 inches below floor level, and the addition of 15,000 gallons would have raised level approximately eight inches.

G. Corrective Action

Procedural inadequacy. Shortly after the incident, the weakness identified in Surveillance Procedure 6.3.4.3 which required two actions with only one set of Operator initials was identified. That portion of the procedure has been revised to provide an initial block for each valve position and power supply.

Lack of mental attention. Operations Procedure 2.0.3 (Control Room Conduct and Manning) will be reviewed and additional guidance given, as appropriate, to control non-operations personnel in the Control Room.

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Procedure not followed. The Operations Supervisor will discuss this event with all Operators and re-emphasize the need to follow procedures.

H. Past Similar Events

Similar events which have occurred in the past and were reported as LERs include:

LER 88-017, dated June 27, 1988, Unplanned Automatic Actuation of Engineered Safety Features Due to Human Errors During Surveillance Testing

Nebraska Public Power District

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TELEPHONE (402) 825-3811

CNSS886192

July 11, 1988

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Gentlemen:

Cooper Nuclear Station Licensee Event Report 88-018 is forwarded as an attachment to this letter.

Sincerely,

J. M. Meacham
Acting Division Manager of
Nuclear Operations

JMM:sg

Attachments

cc: R. D. Martin
L. G. Kuncl
R. E. Wilbur
V. L. Wolstenholm
G. A. Trevors
INPO Records Center
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