Omaha Public Power District
444 South 16th Street Mall
Omaha, Nebraska 68102-2247
402/636-2000

September 22, 1995
LIC-95-0174

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, D.C. 20555

Reference: Docket No. 50-285

Gentlemen:

Subject: Licensee Event Report 95-005 Revision 00 for the Fort Calhoun Station

Please find attached Licensee Event Report (LER) 95-005 Revision 00 dated September 22, 1995. This report is being submitted pursuant to 10 CFR 50.73(a)(2)(iv). If you should have any questions, please contact me.

Sincerely,

for
T. L. Patterson
Division Manager
Nuclear Operations

TLP/epm

Attachment

c: Winston and Strawn
L. J. Callan, NRC Regional Administrator, Region IV
S. D. Bloom, NRC Project Manager
W. C. Walker, NRC Senior Resident Inspector
INPO Records Center

IE22

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Fort Calhoun Station Unit No. 1	05000285	1 OF 4

TITLE (4): Plant Trip Due To Operator Error During Diverse Scram System Testing

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
08	24	95	95	-- 005 --	00	09	22	95		05000
										05000

OPERATING MODE (9): 1

POWER LEVEL (10): 100

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

	20.402(b)		20.405©	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©
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
	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 (Include Area Code)
David J. Bannister, Operations Engineer	(402) 533-6831

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)

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

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On Thursday, August 24, 1995, at 1114 while operating at 100% power (mode 1) the plant experienced a reactor trip and associated diesel generator start. The trip occurred during the performance of IC-PM-DSS-1001, "Diverse Scram System Actuation Relay Operability Test," when a licensed operator inadvertently repositioned the incorrect switch from TEST to NORMAL. Generally, plant systems responded normally to the trip except for diesel generator number 1, which went to full speed (900 RPM) rather than idle speed (500 RPM). The diesel start problem is fully described in LER 95-006.

The Root Cause of the plant trip was inadequate attention to detail. This is a human error that could have been prevented with additional self checking.

Corrective actions include instituting the use of peer-verification for operating equipment from the control room that could cause a significant plant transient and making peer-verification and self-checking a specific item of evaluation during Operations self-assessments.

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

BACKGROUND

The Diverse Scram System (DSS) is provided to comply with 10CFR50.62 that requires all pressurized water reactors to have equipment separate from the Reactor Protection System to initiate a reactor/turbine trip under Anticipated Transient Without Scram (ATWS) conditions. DSS accomplishes this by using diverse, independent components to initiate a reactor trip on an over pressurization of the Reactor Coolant System (RCS). DSS is composed of four independent instrument loops (A/B/C/D), each having an independent pressure transmitter (PT-120 A/B/C/D) and bistable trip unit. The bistable trip unit output contacts are configured into two independent logic matrices (A & B) each using a two-out-of-four logic scheme. Each matrix has an 86 lock-out relay which, when energized, deenergizes the undervoltage trip coils on the control rod clutch power supply breakers. Once open, these breakers interrupt power to the clutch mechanisms resulting in the control rods dropping into the core, thereby shutting down the reactor. An automatic trip signal is generated if RCS pressure, as sensed on two-of-four DSS pressure transmitters, reaches 2450 psia (normal RCS pressure is 2100 psia). A manual DSS trip can be initiated from either channel using either of two (2), A/TS-DSS & B/TS-DSS, two position (NORMAL/TRIP) manual trip switches.

DSS was designed to permit on-line testing of the individual channels and the matrix actuation relays. To prevent a reactor trip during testing a three position (BYPASS/NORMAL/TEST) DSS TEST OR BYPASS KEY SWITCH (A1/TS-DSS & B1/TS-DSS) is provided for each channel. In the BYPASS position, the 86 lock-out relay is prevented from being energized (tripped). In TEST, blocking relays are energized to close contacts which prevent de-energization of the undervoltage coils of the clutch power supply breakers. The key lock switches have only one key, which is removable in the NORMAL position, therefore only one channel may be bypassed/tested at a time.

Currently DSS is tested during power operation on a monthly basis using preventative maintenance procedure IC-PM-DSS-1001, "Diverse Scram System Actuation Relay Operability Test." The test verifies the operability of the individual channels, including the bistables, pre-trips, trips, and control functions. The test also verifies manual trip capability, as well as matrix actuation relay and matrix blocking relay operability. The test is broken into seven sections. Section one is a switch line-up performed by Operations personnel to ensure the DSS is properly aligned for the test. Sections two through five are the individual channel (A/B/C/D) actuation relay operability tests. They are primarily done by Instrument and Control (I&C) Technicians. Sections six and seven are the matrix channel (A & B) manual actuation relay tests, and are done by Operations personnel.

EVENT DESCRIPTION

On Thursday morning, August 24, 1995, with the station at 100% power (Mode 1), one licensed operator and two I&C technicians were performing IC-PM-DSS-1001, "Diverse Scram System Actuation Relay Operability Test." The first six sections of the test had been completed satisfactorily. Section seven, "DSS Matrix Channel B Manual Trip Actuation Relay Test" was being performed to complete the test. The licensed operator performing the test had just completed step 6.7.4 of the test. During step

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Fort Calhoun Station Unit No. 1	05000285	95	-- 005 --	00	3 OF 4

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

6.7.4 the operator had correctly placed the DSS MANUAL TRIP SWITCH, B/TS-DSS, in the TRIP position. This action caused the DSS lock-out relay 86B/DSS to go to the tripped position, as expected. The actuation of the lock-out relay would have caused the reactor to trip, however, the CHAN B DSS TEST OR BYPASS SW B1/TS-DSS had been correctly placed in the TEST position in step 6.7.2 of the procedure. The next step in the procedure sequence (6.7.5) was to place switch DSS MANUAL TRIP SWITCH B/TS-DSS in the NORMAL position. This would have allowed resetting the DSS lock-out relay and recovering from the test. The operator inadvertently repositioned the key operated switch CHAN B DSS TEST OR BYPASS SW B1/TS-DSS from the TEST position to the NORMAL position. This action combined with the lock-out being in the tripped position, caused the reactor to trip at 11:14 am.

The plant systems responded as expected for an uncomplicated reactor trip with the exception of emergency diesel generator number 1 (DG-1) going to full speed (900 RPM) rather than idle speed as expected (The diesel generator problem is fully described in LER 95-006.). The licensed operators in the control room carried out Emergency Operating Procedure - 00 (EOP-00), "Standard Post Trip Actions," and upon diagnosing the event as an uncomplicated reactor trip transitioned to Emergency Operating Procedure - 01 (EOP-01), "Reactor Trip Recovery".

SAFETY ASSESSMENT

This event is not considered significant with regard to nuclear safety. Plant personnel responded appropriately resulting in a safe reactor trip recovery. With the exception of DG-1 going to full speed, all plant systems actuated as designed. This event, however, did result in an unplanned automatic reactor trip from full power and start of both emergency diesel generators (which is reportable under 10CFR50.73(a)(2)(iv)). This unplanned shutdown reduced plant reliability and availability as the station was unable to return to power operation for approximately two days.

Upon further evaluation, the full speed start of DG-1 resulted in a violation of FCS Technical Specifications section, 2.7.2.j. The DG-1 event is being reported separately under LER 95-006.

CONCLUSIONS

A Root Cause Analysis (RCA) determined that the root cause of this event was inadequate attention to detail which resulted in a human error that could have been prevented with additional self-checking.

The operator assigned to the test was qualified and experienced and had previously performed the procedure. The operator had just completed the test on the "A" channel and recalled placing his hand on each switch and verifying the switch as correct prior to manipulation. The operator could not recall if he used the same self-checking technique on the "B" channel when the wrong switch was manipulated. The operator performing the test was not distracted or under any time constraints.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Fort Calhoun Station Unit No. 1	05000285	95	- 005 -	00	4 OF 4

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

As a part of the RCA, a review for generic implications was conducted. Incident Reports assigned to the Operations department and coded as human error were reviewed to determine if an excessive number of human performance errors existed. The review noted that in 1995 there were a number of human performance related incident reports. As a result of this review the Operations Department will be instituting actions to reduce the number of these problems (see Corrective Actions).

CORRECTIVE ACTIONS

The following corrective actions have been or will be completed:

1. The Control Room Operators are now required to verbalize pertinent label information, as an additional self-checking technique, prior to operating equipment from the Control Room.

2. Peer-verification has been implemented, for Control Room Operators, prior to operating equipment from the Control Room that could immediately cause a significant plant transient.

3. Operations Policies and Directives OPD-4-15, "Conduct of OPEP Effectiveness/Operations Self Assessment Policy," has been revised to include peer verification.

4. During the next Operations Self Assessment an evaluation of self-checking and peer-verification will be included. This assessment will be completed by March 31, 1996.

5. Station management will conduct an evaluation of the Operations Performance Enhancement Program (OPEP) to determine its effectiveness and make any needed corrections to the program. The evaluation and a schedule for corrective actions will be completed by December 31, 1995.

SIMILAR EVENTS

LER 92-005 discusses a previous event where a reactor trip signal was generated due to not bypassing a signal correctly. That event occurred during a plant shutdown and cooldown.