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)
Limerick Generating Station, Unit 2	05000 353	1 OF 6

TITLE (4) Reactor SCRAM and Actuation of Various Engineered Safety Features Resulting from a Relay Coil Failure and Inappropriate Action By a Licensed Reactor Operator.

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 NAME	DOCKET NUMBER
										05000
10	19	94	94	-- 010 --	01	10	13	95	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 1

POWER LEVEL (10): 093

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (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)	X	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 (Include Area Code)
J. L. Kantner, Manager, Experience Assessment, LGS	(610) 718-3400

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 15 single-spaced typewritten lines) (16)

On 10/19/94, a licensed operator (LO) performing a surveillance test inadvertently de-energized the D24 4KV Safeguard Bus. Immediately following the de-energization, the Reactor Feedwater Pump (RFP) minimum flow control valves fully opened. A reactor SCRAM occurred on low reactor water level, as designed. During the transient, reactor water level reached the low-low level setpoint, and the Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection systems received initiation signals. The RCIC system injected to recover reactor water level. Actuation of the Primary Containment and Reactor Vessel Isolation Control System also occurred, as designed, as a result of the vessel low-low water level signal. The reactor shutdown was accomplished with no abnormalities. The reactor SCRAM was caused by the failure of the N-relay coil in the RFP minimum flow control valve control power supply transfer logic. The cause of the N-relay coil failure was insulation breakdown, and is considered isolated. De-energization of the D24 Safeguard Bus occurred due to an inappropriate action by a LO when securing the D24 Emergency Diesel Generator following testing. Corrective actions included a failure analysis of the relay coil, an evaluation of the normal to alternate power supply transfer circuits, and discipline of the involved LO.

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Limerick Generating Station, Unit 2	05000353	94	-- 010 --	01	2 OF 6

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

Unit Conditions Prior to the Event:

Unit 2 was in Operational Condition (OPCON) 1 (Power Operation) at 93% power level in end of cycle coastdown. Operations personnel were performing monthly Surveillance Test (ST) procedure ST-6-092-318-2, "D24 Diesel Generator Fast Start Operability Test Run."

There were no structures, systems, or components out of service that contributed to this event.

Description of the Event:

On October 19, 1994, at 12:17:45 hours, a Licensed Operator (LO) performing procedure ST-6-092-318-2 inadvertently de-energized the D24 4KV Safeguard Bus. Immediately following the de-energization, the Reactor Feedwater Pump (RFP, EIIS:SJ) minimum flow control valves fully opened. This diverted some feedwater flow to the main condenser and caused a reduction in reactor vessel water level. The RFP controls raised RFP speeds in an attempt to maintain reactor level. The 'C' RFP tripped due to low feedwater pump suction pressure. At 12:18:21 hours, a reactor SCRAM occurred on low reactor water level (+12.5 inches, the zero reference point being 161 inches above the top of active fuel) as designed. During the transient, reactor water level reached the low-low level setpoint (-38 inches) at 12:18:28 hours, and the Reactor Core Isolation Cooling (RCIC, EIIS:BN) and High Pressure Coolant Injection (HPCI, EIIS:BJ) systems received initiation signals. The RCIC system injected water into the Reactor Pressure Vessel and, combined with the remaining feedwater flow, recovered reactor water level to +12.5 inches by 1219 hours. As reactor water level increased and approached +12.5 inches, Operators secured the HPCI system in accordance with station procedures, prior to automatic injection. Reactor water level reached a minimum of -64 inches during the transient. All automatic safety functions operated as designed.

Actuation of Primary Containment and Reactor Vessel Isolation Control System (PCRVICS, EIIS:JM) isolations, Engineered Safety Features (ESFs), also occurred as designed as a result of the vessel low-low water level signal. The PCRVICS actuations resulted in the isolation of the following Unit 2 systems:

- Main Steam and Reactor Sample Lines,
- Residual Heat Removal (EIIS:BO) Heat Exchanger Vacuum Breaker Lines,
- Reactor Water Cleanup,
- Primary Containment and Purge Supply and Exhaust Lines,
- Primary Containment Sampling and Recombiner Lines,
- Primary Containment TIP Purge Supply,
- Drywell Sump, Suppression Pool Cleanup and TIPS, and
- Instrument Gas Blocks and Vents.

The following additional ESFs initiated as designed due to the PCRVICS actuations. The Reactor Enclosure (RE) Heating, Ventilation and Air Conditioning (HVAC) system isolated. The 'A' and 'B' trains of the Standby Gas Treatment System (SGTS, EIIS:BM), a common plant system, and the 'B' train of Unit 2 Reactor Enclosure Recirculation System (RERS, EIIS:VA), automatically initiated thus completing the RE Secondary Containment isolation.

The Transient Response Implementation Plan (TRIP) procedures T-101, "RPV Control," and T-99, "Post SCRAM Restoration," were executed by Main Control Room (MCR) personnel following the reactor SCRAM. The reactor shutdown was accomplished with no abnormalities. All control rods fully inserted following the reactor SCRAM. Reactor coolant level was restored to normal level using the RCIC and feedwater systems. General Plant (GP) procedure GP-3, "Normal Plant Shutdown," was executed to continue with normal shutdown activities. Procedure GP-8, "Primary and Secondary Containment Isolation Verification and Reset," was executed to reset the PCRVICS isolation signal. Following recovery from the SCRAM and a review of the event, Unit 2 was taken critical on October 21, 1994, at 0356 hours.

A one hour notification was made to the NRC at 1314 hours, on October 19, 1994, in accordance with the requirements of 10CFR50.72(b)(1)(iv) since this event resulted in a valid initiation of the HPCI system, an Emergency Core Cooling System. Additionally, the notification satisfied the four hour notification requirement of 10CFR50.72(b)(2)(ii), which was required since this event resulted in automatic actuations of the Reactor Protection System and ESFs. This LER is being submitted in accordance with the requirements of 10CFR50.73(a)(2)(iv).

In addition, TS 3/4.7.3, "RCIC System Limiting Conditions for Operation," Action 3.7.3.b, requires that in the event the RCIC system is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. This was the ninth RCIC system actuation cycle since the original startup of Unit 2. This LER provides the information required to be in the Special Report.

Analysis of the Event:

The reactor SCRAM occurred automatically and all control rods fully inserted as designed. MCR Operations personnel successfully controlled shutdown of the plant using the appropriate station procedures. No reactor vessel Main Steam Relief Valves lifted, and the RCIC system injected water into the reactor to restore reactor vessel level. The actuation of the PCRVICS functioned as designed in response to the reactor vessel low-low water level signal. There was no release of radioactive materials to the environment as a result of this event. The HPCI system was not needed to restore reactor vessel water level. The other three (3) Unit 2 4KV Safeguard Buses, the associated three (3) Emergency Diesel Generators (EDG, EIIS:EK), and the two (2) offsite sources were operable throughout the event to provide power to assure safe shutdown capability.

Cause of the Event:

The cause of the event was a relay coil failure resulting from insulation breakdown combined with an inappropriate action by a licensed operator. The decrease in reactor water level was caused by a failure of the N-relay coil in the non-safety related RFP minimum flow control valve control power supply transfer logic. The relay failure prevented automatic transfer of the control valve power supply to the D23 Safeguard Bus when the D24 Safeguard Bus was inadvertently de-energized. An automatic transfer would have prevented the disruption in feedwater flow and the SCRAM.

De-energization of D24 Safeguard Bus occurred due to an inappropriate action by a MCR LO when securing the D24 EDG following surveillance testing. This action requires manipulation of several handswitches on MCR panel 2DC661 (See Attachment 1, "D24 Diesel Generator Panel 2DC661

Arrangement"). The procedure directed the LO to open the D24 Generator Output Breaker, then place the D24 Control Switch to "STOP." Instead, the LO opened the 101-D24 Bus Breaker and the D24 Generator Output Breaker Switch, de-energizing the D24 Safeguard Bus. The LO did not verify that he was operating the correct switches. Had the procedure steps been performed as written, the bus would have remained energized.

Corrective Actions:

1. A failure analysis of the N-relay coil and the remaining 3 Unit 1 and Unit 2 similar relay coils was performed. The analysis identified that the subject Unit 2 relay coil failed as a result of insulation breakdown. The results for the 3 remaining relay coils indicated that the coils functioned properly, and no signs of failure, fatigue, or corner stress existed. Based on these conclusions, this relay coil failure is considered an isolated occurrence and no generic concerns exist with these particular relay coils.

2. An evaluation was performed to determine the appropriate testing or preventative maintenance (PM) inspection requirements for the normal to alternate power supply transfer circuits utilized at the Limerick Generating Station (LGS), including the circuit involved in this event. Appropriate PMs have been developed and scheduled as a result of this evaluation.

3. The LO was disciplined for his failure to conform to the management expectation to ensure that he manipulates the correct equipment, in accordance with approved procedures.

4. Long term initiatives to enhance human performance and self checking behaviors of plant personnel are in place and continue.

Previous Similar Occurrences:

LGS Unit 1 LER 1-93-011 also reported a SCRAM where a contributing cause was a failure of a power supply relay. Corrective actions from that event included an investigation into similar safety related circuits. The relay coil failure in this event involved a dissimilar non-safety related power supply transfer circuit, and therefore the corrective actions from LER 1-93-011 would not have prevented the event reported in this LER.

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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Attachment 1
D24 Diesel Generator Panel 2DC661 Arrangement