Entergy

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Michael A. Balduzzi
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November 3, 2003

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

Subject	Entergy Nuclear Operations, Inc. Pilgrim Nuclear Power Station Docket No. 50-293 License No. DPR-35 Licensee Event Report 2003-005-00
Letter Number:	2.03.127

Dear Sir:

The enclosed Licensee Event Report (LER) 2003-005-00, "HPCI and RCIC Systems Inoperable due to De-energized Train 'A' Electrical Loads Caused by Circuit Breaker Current Transformer Malfunction," is submitted in accordance with 10 CFR 50.73

This letter contains no commitments.

Please feel free to contact me if there are any questions regarding this subject.

Sincerely,

Michael A. Balduzzi

Michael A. Balduzzi

DWE/dm

cc: Mr. Hubert J. Miller
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INPO Records

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1. NRC Form 366 U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104
LICENSEE EVENT REPORT (LER) (See reverse for number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), 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. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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TITLE (4)
HPCI and RCIC Systems Inoperable due to De-energized Train 'A' Electrical Loads Caused by Circuit Breaker Current Transformer Malfunction

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 N/A	DOCKET NUMBER 05000
09	06	2003	2003	005	00	11	03	2003	FACILITY NAME N/A	DOCKET NUMBER 05000

OPERATING MODE (9)	N
POWER LEVEL (10)	100

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

	20.2201(b)		22.2203(a)(3)(i)		50.73(a)(2)(i)(C)		50.73(a)(2)(vii)
	22.2202(d)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(A)
	20.2203(a)(1)		20.2203(a)(4)		50.73(a)(2)(ii)(B)		50.73(a)(2)(viii)(B)
	20.2203(a)(2)(i)		50.36(3)(1)(i)(A)		50.73(a)(2)(iii)		50.73(a)(2)(ix)(A)
	20.2203(a)(2)(ii)		50.36(3)(1)(ii)(A)		50.73(a)(2)(iv)(A)		50.73(a)(2)(x)
	20.2203(a)(2)(iii)		50.36(c)(2)		50.73(a)(2)(v)(A)		73.71(a)(4)
	20.2203(a)(2)(iv)		50.46(a)(3)(ii)	X	50.73(a)(2)(v)(B)		73.71(a)(5)
	20.2203(a)(2)(v)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(C)		OTHER
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(B)	X	50.73(a)(2)(v)(D)		Specify in Abstract below or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Bryan Ford – Licensing Manager	(508) 830-8403

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	ED	BKR	G080	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 15 single-spaced typewritten lines) (16)

On September 6, 2003 at 1151 hours, the safety-related train 'A' 480-volt load center and the related motor control centers (MCCs) powered from the load center de-energized. The event resulted in several systems including the high pressure coolant injection and the reactor core isolation cooling systems becoming inoperable. The train 'B' 480-volt load center and related MCCs were unaffected. The affected load center and related MCCs were re-energized by 0022 hours and the affected systems were returned to service by 1000 hours on September 7, 2003.

The direct cause of the event was the unplanned trip of the circuit breaker that powers the load center. The root cause investigation revealed the circuit breaker tripped due to a malfunction of one of the breaker's three current transformers.

Corrective action included the replacement of the circuit breaker. Corrective action planned includes analysis of the current transformer malfunction and related actions to preclude recurrence.

The event posed no threat to public health and safety.

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

BACKGROUND

The preferred offsite power source for Pilgrim Station is the 345 kV transmission lines 342 and/or 355 via the Startup Transformer. The secondary offsite power source is the 23 kV distribution system via the Shutdown Transformer.

The Pilgrim Station 4.16 kV auxiliary power distribution system (APDS) provides power to nonsafety-related and safety-related loads. The nonsafety-related electrical loads are powered from 4.16 kV Buses A1 through A4 and the respective 480-volt load centers and motor control centers (MCCs). The safety-related and certain nonsafety-related electrical loads are powered from 4.16 kV Buses A5 and/or A6 and the respective 480-volt load centers and MCCs.

Safety-related 480-volt train 'A' load center B1 is powered from 4.16 kV Bus A5 via a step-down transformer (X-21) and circuit breaker (B101). Safety-related 480-volt train 'B' load center B2 is powered from 4.16 kV Bus A6. Safety-related 480-volt swing load center B6 is normally powered from B1 and is designed to automatically transfer from B1 to B2 if B1 becomes de-energized. Located at the end of this report is a figure that depicts a simplified single-line diagram of the APDS including load centers B1, B2, and B6.

Just prior to the event the following conditions existed. The preferred and secondary offsite power sources were energized, the switchyard ringbus was intact, the Emergency Diesel Generators and the Station Blackout Diesel Generator were in standby service. The Main Transformer was energized by the Main Generator. The APDS was being powered from the Startup Transformer. The Unit Auxiliary Transformer was removed from service for replacement. Swing load center B6 was being powered from load center B1. The reactor was operating at 100% power (2028 MWt) with the reactor mode selector switch in the RUN position. The reactor vessel pressure was normal, about 1035 psig, with the reactor water at the saturation temperature for that pressure. The reactor water level was normal, at about +28" (narrow range).

EVENT DESCRIPTION

On September 6, 2003 at 1151 hours, safety-related 480-volt train 'A' load center B1 and related motor control centers (MCC) powered from load center B1 de-energized. This was the result of the unplanned trip of the safety-related 480-volt AC circuit breaker (B101) that powers load center B1, and resulted in train 'A' safety systems becoming inoperable, the high pressure coolant injection (HPCI) system being made inoperable, and the reactor core isolation cooling (RCIC) system being declared inoperable.

The de-energizing of load center B1 resulted in the following:

The automatic transfer of the source of power to safety-related swing load center B6 from load center B1 to load center B2. The transfer resulted in the brief loss of power to the swing 120-volt AC instrument power supply Panel Y1 that included the transfer of reactor pressure control from the electric pressure regulator to the mechanical pressure regulator, and loss of feedwater heating resulting in a brief increase in reactor power that was within analyzed limits.

The loss of electrical power to safety-related MCCs B15 and B17, and resulted in the loss of AC power to or inoperability of the following safety-related systems:

- The normally open high pressure coolant injection (HPCI) system turbine steam primary containment isolation valve (MO-2301-4). The in-series containment isolation valve was manually closed and de-energized consistent with primary containment Technical Specifications, and this action resulted in the HPCI system to be inoperable.

- Emergency diesel generator (EDG) 'A' and train 'A' of the residual heat removal (RHR), core spray system, standby liquid control (SBLC) system, standby gas treatment (SBGT) system, and the control room high efficiency air filtration (CRHEAF) system. The operability of EDG 'B' and train 'B' of the RHR, core spray, SBLC, SBGT, and CRHEAF systems were not affected.

- The area coolers in certain reactor building areas including the reactor core isolation cooling (RCIC) system turbine-pump. The RCIC system was declared inoperable but was maintained in a standby condition and was available for its safety function. The coolers in the areas containing the RHR train 'B' pumps and heat exchanger and core spray train 'B' pump, high pressure coolant injection (HPCI) system turbine-pump, and control rod drive pumps were not affected.

- The reactor building closed cooling water (RBCCW) system loop 'A' pumps ('A', 'B', and 'C') and related loop A' valves. The RBCCW loop 'B' was not affected and the valves that cross-tie RBCCW system loops 'A' and 'B' were opened in accordance with procedure. This action restored forced cooling water flow to RBCCW loop 'A' heat loads via the RBCCW loop 'B' pumps.

- The salt service water (SSW) system train 'A' pumps. The SSW system train B' pumps were not affected. The SSW swing pump 'C' was manually started because the SSW train 'A' pumps were inoperable.

- The de-energizing of 120 volt safeguards panels Y3 and Y31 including the normally energized relays that control the inboard primary containment isolation valves including the Group 6/reactor water cleanup (RWCU) isolation valve MO-1201-2. Valve MO-1201-2 closed automatically as designed and resulted in the loss of the nonsafety-related function of the RWCU system. The outboard Group 6/RWCU isolation valves were manually closed. A manual actuation of the reactor building isolation control system (RBIS) was initiated in accordance with procedure to restore reactor building ventilation.

- The loss of power to the recirculation system motor-generator (MG) set 'A' oil pumps, and resulted in the automatic trip and lockout of the MG set/pump 'A' and consequent decrease in reactor power. Reactor power stabilized at about 75% (single recirculation loop operation).

Additional licensed operator response included the entry into emergency and abnormal procedures including those for a loss of feedwater heating, loss of power to panels Y1 and Y3/31, loss of RBCCW, and loss of one SSW train.

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After the initial investigation and replacement of the circuit breaker in the B101 application, load center B1 was re-energized at 2351 hours on September 6, 2003. On September 7, 2003 at 0009 hours, power was restored to panels Y3/Y31. MCCs B15 and B17 and related loads were re-energized by 0022 hours on September 7, 2003. The affected systems were returned to service by 1000 hours on September 7, 2003.

The NRC Operations Center was notified of the event in accordance with 10 CFR 50.72 at 1711 hours on September 6, 2003.

CAUSE

The direct cause of the event was the unplanned trip of the circuit breaker in the B101 application. The circuit breaker installed in the B101 application at the time of the event was a General Electric type AK-50 circuit breaker, serial number 256A9428-207-3EL, equipped with a Model RMS-9 micro-versa trip unit.

Troubleshooting was performed to determine if an electrical fault had occurred on load center B1. This action revealed that a fault had not occurred on load center B1. Insulation resistance testing of load center B1 was performed with satisfactory results.

Investigation concluded there was a malfunction in one of the three current transformers (CTs) that are part of the circuit breaker. The CTs sense current on the electrical phases of the breaker. Vendor support has been requested to identify the reason for the CT malfunction.

CORRECTIVE ACTION

Corrective action taken included the following:

- Replacement of the circuit breaker in the B101 application. Prior to the installation of the replacement circuit breaker, visual inspections of the replacement breaker were performed with satisfactory results. The breaker was cycled 15 to 20 times and overcurrent testing was performed, all with satisfactory results.

- After the replacement breaker was installed, load center B1 was energized and the breaker functioned as designed.

- In addition to the initial investigation and troubleshooting, the circuit breaker that caused the event was sent to a laboratory for investigation and analysis of the CT malfunction.

Corrective actions planned will be tracked in the corrective action program, and include the review of the laboratory report of the investigation and analysis of the CT malfunction. The focus of this action is to determine corrective actions to preclude recurrence.

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SAFETY CONSEQUENCES

This event posed no threat to public health and safety.

The core standby cooling systems (CSCS) consist of the HPCI system, automatic depressurization system (ADS), core spray system, and the RHR system (LPCI mode). Although not a CSCS system, the RCIC system is capable of providing high pressure core cooling similar to the HPCI system. The HPCI system is designed to provide high pressure core cooling in the event of a loss of coolant accident (LOCA) which does not result in a rapid depressurization of the reactor vessel. If a LOCA signal had been received while the HPCI system was inoperable, an automatic (or manual) actuation of the ADS would function to reduce the reactor vessel pressure for low pressure core cooling provided independently by the core spray system and RHR/LPCI mode.

EDG 'A' and train 'A' of the SBGT, CRHEAF, SBLC, core spray, RHR, RBCCW and SSW systems became inoperable as a result of the de-energizing of the respective equipment required for operability. EDG 'B' and train 'B' of these systems were unaffected by the event. The Pilgrim Station LOCA analysis has demonstrated the combination of core spray and RHR/LPCI from one division (e.g. operable train 'B') are sufficient to provide core cooling.

The HPCI system was made inoperable by the manual closing and de-energizing of the HPCI turbine steam primary containment isolation valve (MO-2301-5) because the in-series containment isolation valve (MO-2301-4) was open and de-energized (MCC B15 de-energized). This action was taken consistent with Technical Specifications for an inoperable automatic containment isolation valve. If necessary, the HPCI system could have been returned to service for its safety function through simple licensed operator actions consisting of reenergizing valve MO-2301-5 and the slight, local opening of valve MO-2301-5 to slowly pressurize the downstream HPCI turbine steam supply piping. These actions were discussed during a briefing of the on-shift licensed operators including the assigned operator who was performing licensed duties during the period when the HPCI system was inoperable and who was dedicated to perform these actions if necessary.

Moreover, although the RCIC system (non-CSCS) was declared inoperable because of the loss of AC power to the RCIC system quadrant area coolers' fan motors, the RCIC system was maintained in a standby condition and was available for service if necessary. The RCIC system was declared inoperable because of the loss of the function of the coolers in the RCIC turbine-pump area. There is no analysis for determining the length of time the system can operate without exceeding area temperatures that could impact RCIC system operability but the system would be expected to operate for several hours without the area cooling.

REPORTABILITY

This report was submitted in accordance with 10 CFR 50.73(a)(2)(v)(B) and (D) because the HPCI system was made inoperable and the RCIC system was declared inoperable.

SIMILARITY TO PREVIOUS EVENTS

A review for similarity was conducted of Pilgrim Station Licensee Event Reports (LERs) submitted since 1984. The focus of the review was on events or conditions involving an unplanned trip of a circuit breaker or similar cause. The review identified the following.

The unplanned de-energizing of load center B1 due to the unexplained trip of 4.16 kV switchgear breaker A508 that occurred while shut down was reported in LER 87-024-00. The unplanned de-energizing of load center B2 due to the unplanned loss of power to Bus A6 that occurred during power operation was reported in LER 91-005-01. Neither event involved the failure of a current transformer.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

COMPONENTS	CODE
Bus (load center B1)	BU
Breaker (B101)	BKR
Control center, motor	MCC
Fan coil unit (area cooler)	FCU
Heater, air	EHTR
Motor	MO
Panel	PL
Pump	P
Valve, electrically operated (MO)	20

SYSTEMS	CODE
Component cooling system (RBCCW, TBCCW)	CC
Control complex environmental control system (CRHEAF)	VI
Control rod drive system	AA
Core spray system	BM
Emergency onsite power system (EDG'A')	EK
Essential service water system (SSW)	BI
Feedwater system	SJ
High pressure coolant injection system (HPCI)	BJ
Low voltage power system – Class 1E	ED
Reactor core isolation cooling system (RCIC)	BN
Reactor recirculation system	AD
Reactor water cleanup system (RWCU)	CE
Residual heat removal system (RHR)	BO
Standby gas treatment system (SBGT)	BH
Standby liquid control system (SBLC)	BR

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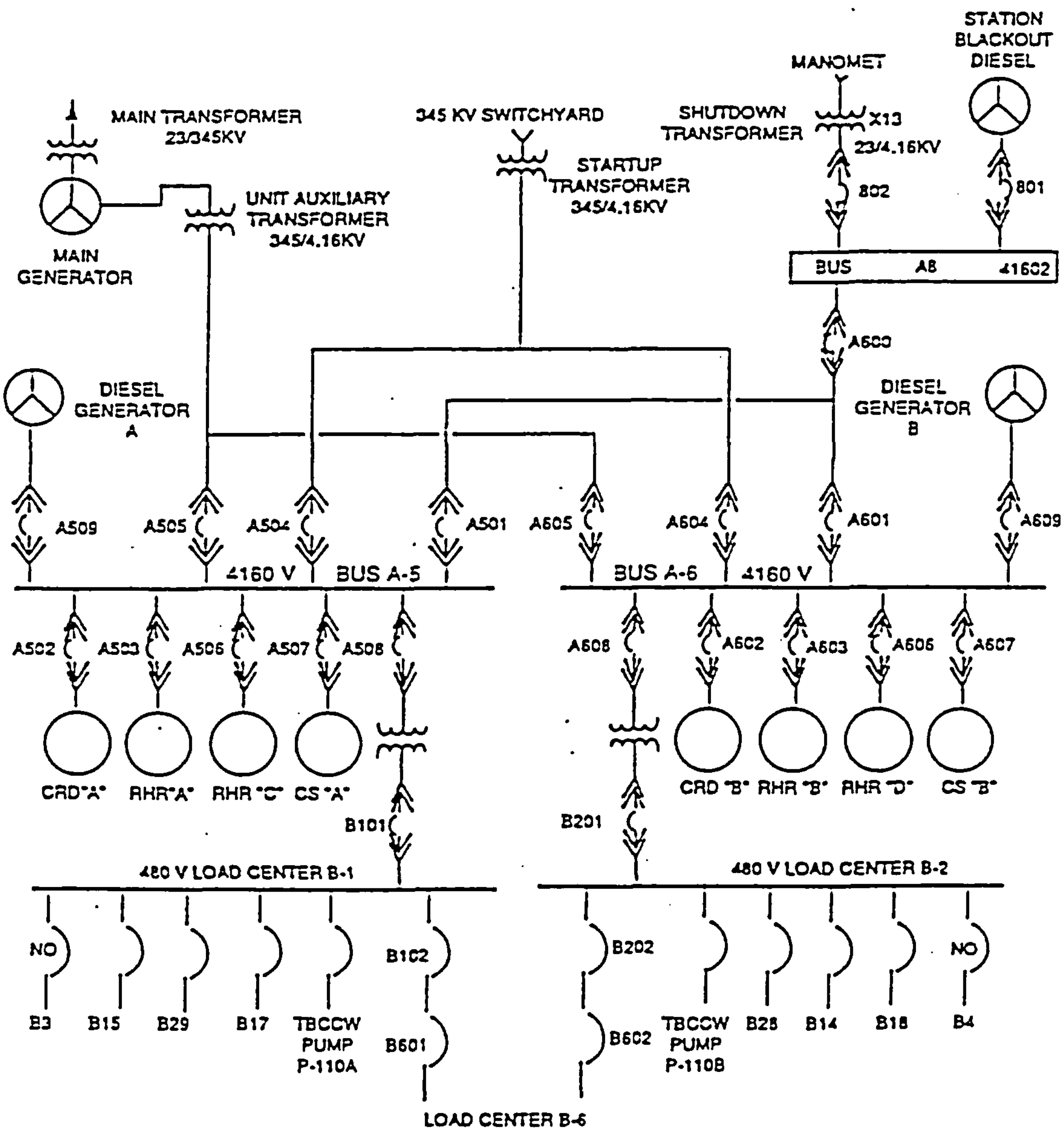

EMERGENCY AC DISTRIBUTION

Figure of APDS