10 CFR 50.73

BOSTON EDISON
Pilgrim Nuclear Power Station
Rocky Hill Road
Plymouth, Massachusetts 02360

Ralph G. Bird
Senior Vice President — Nuclear

June 12, 1990
BECo Ltr. 90-074

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

Docket No. 50-293
License No. DPR-35

Dear Sir:

The enclosed Licensee Event Report (LER) 90-008-00, "Automatic Scram Resulting From Load Rejection at Full Power", is submitted in accordance with 10 CFR Part 50.73.

Please do not hesitate to contact me if there are any questions regarding this report.

R. G. Bird

DWE/bal

Enclosure: LER 90-008-00

cc: Mr. Thomas T. Martin
Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Rd.
King of Prussia, PA 19406

Sr. NRC Resident Inspector - Pilgrim Station

Standard BECo LER Distribution

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

LICENSEE EVENT REPORT (LER)

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TITLE (4)
Automatic Scram Resulting From Load Rejection At Full Power

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)
									N/A	0 5 0 0 0
0 5	1 3	9 0	9 0	0 0 8	0 0	0 6	1 2	9 0	N/A	0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 1 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)
Douglas W. Ellis - Senior Compliance Engineer	5 0 8 7 4 7 - 8 1 6 0

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	E L	1 9 2	W 1 2 0	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 fifteen single-space typewritten lines) (16)

On May 13, 1990 at 1603 hours, an automatic scram resulting from a load rejection occurred while at 100 percent reactor power. The load rejection included a trip of the Generator Field Breaker, actuation of the Turbine mechanical hydraulic control speed governor, closure of the four Turbine Control Valves and opening of the three Bypass Valves, and the brief actuation of the Main Steam/Target Rock two-stage relief valves at approximately 1100 psig (low end of the 1115 psig setpoint range including tolerance).

The load rejection was caused by a momentary fault on the offsite 345 KV transmission system. The Generator's loss-of-field relay (240) detected the fault and immediately tripped the Generator without an expected (inherent) 15 cycle time delay because one of it's components, the telephone relay ('X') coil, was defective. The relay (240) was last calibrated and functionally tested on October 26, 1989. At that time, the operation of the ('X') coil was tested in accordance with the vendor manual. The relay's time delay was built-in and not adjustable, and was not required to be timed. The relay was installed during plant construction (c. 1972). The cause for the open coil is being investigated but is believed to be random or age related failure. The relay is the only one of its type (Westinghouse type KLF-1) installed at Pilgrim Station and was replaced with another KLF-1 relay having an adjustable time delay. The (240) relay's calibration sheet was revised to include a calibration of the adjustable time delay.

This event occurred with the reactor mode selector switch in the RUN position. The Reactor Vessel (RV) pressure was initially at 1035 psig with RV water temperature at 548 degrees Fahrenheit. This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv) and this event posed no threat to the public health and safety.

EVENT DESCRIPTION

On May 13, 1990 at 1603 hours, an unplanned automatic Reactor Protection System (RPS) scram signal and reactor scram occurred while at 100 percent reactor power. The scram signal occurred as a result of a load rejection that included a trip of the Turbine-Generator.

The trip of the Generator lockout relay (286-1) resulted in the following designed responses:

- Automatic opening of the Generator Field Breaker (41M).
- Automatic opening of the 345 KV switchyard air circuit breakers ACB-104 (352-4) and ACB-105 (352-5).
- Automatic transfer of the source of 4160 VAC power for the Auxiliary Power Distribution System (APDS) from the Unit Auxiliary Transformer (UAT) to the Startup Transformer (SUT).
- Automatic trip of the Turbine master trip solenoid (MTS-1) that resulted in the closing of the Turbine Stop Valves and Combined Intermediate Valves, and the trip of the Turbine lockout relay (286-2).

Concurrently, the Generator trip resulted in a momentary increase in Turbine speed that was caused by the sudden mismatch in Generator load (zero percent) and Turbine power (100 percent). The rapid acceleration actuated the mechanical speed governor in the Mechanical Hydraulic Control (MHC) portion of the Turbine Control System, and an adjustment in the mechanical linkage connected to the acceleration relay. The actuation of the acceleration relay resulted in the following designed responses:

- Fast closure of the 4 (four) Turbine Control Valves and the subsequent opening of the 3 (three) Turbine Bypass Valves.
- Loss of oil pressure to pressure switches (PS-37/38/39/40) that resulted in the RPS scram signal (Turbine Control Valve Fast Closure).

The Main Steam/RV pressure increased as a result of the fast closure of the Turbine Control Valves. The opening of the Bypass Valves, controlled by the pressure regulator, mitigates the pressure transient; however, the pressure increased due to the 25 percent total bypass capacity of the Bypass Valves. The pressure, initially at 1035 psig, increased to approximately 1100 psig and resulted in the automatic actuation of the Main Steam/Target Rock two-stage relief valves RV-203-3A (s/n 1040), RV-203-3B (s/n 1048), and RV-203-3C (s/n 1046). Relief valve RV-203-3D (s/n 1025) did not actuate.

As expected, the RV water level decreased in response to the scram because of shrink (i.e., decrease in the void fraction in the RV water). The RV water level eventually decreased to approximately -10 inches (narrow range level). The decrease in RV water level, to less than the low RV water level setpoint (calibrated at approximately +12 inches) resulted in automatic actuations of the Primary Containment Isolation Control System (PCIS) and Reactor Building Isolation Control System (RBIS).

The PCIS actuation resulted in the following designed responses:

- Automatic closing of the inboard and outboard Primary Containment System (PCS)/Reactor Water Sample isolation valves AO-220-44 and -45.
- Automatic closing of the inboard and outboard PCS Group 2 (two) isolation valves that were open.
- The PCS Group 3 (three)/Residual Heat Removal System isolation valves, in the closed position, remained closed.
- Automatic closing of the inboard and outboard PCS Group 6 (six)/Reactor Water Cleanup (RWCU) System isolation valves and a temporary interruption in RWCU System operation.

The RBIS actuation resulted in the automatic closing of the Reactor Building/Secondary Containment System (SCS) supply and exhaust ventilation dampers (Trains 'A' and 'B'), and the automatic start of Trains 'A' and 'B' of the SCS/Standby Gas Treatment System (SGTS).

Initial Control Room operator response was orderly and included the following. The reactor mode selector switch was moved from the RUN position to the REFUEL position and the reactor feedpumps were tripped in accordance with procedure 2.1.6, "Reactor Scram". Emergency Operating Procedure (EOP)-01, "RPV Control", was initiated when the RV water level decreased to less than +9 inches (narrow range) and was terminated when the RV water level increased to greater than +9 inches. Meanwhile, the High Pressure Coolant Injection System (HPCIS) was manually started in the full flow test mode as a precautionary pressure control measure and in accordance with the guidance provided in EOP-01. The Residual Heat Removal System (RHRS) loop 'A' (pump 'A') was placed in the Suppression Pool Cooling (SPC) mode in accordance with procedure 2.2.86, "Residual Heat Removal", because of the heat addition from the steam discharged into the Suppression Pool via the discharge piping from the relief valves and HPCIS turbine. The Suppression Pool temperature was logged in accordance with procedure 2.1.19, "Suppression Chamber Temperatures". Procedures 2.1.5 Attachment 1, "Shutdown/Cooldown Checklist", and 2.1.7 Attachment 1, "RPV Temperature and Pressure Checklist", were initiated. The PCIS circuitry was reset and the RWCU System was returned to service. The RBIS circuitry was reset, the SGTS was returned to normal standby service and the Reactor Building ventilation system was returned to normal service.

The NRC Operations Center was notified of the event in accordance with 10 CFR 50.72 on May 13, 1990 at 1734 hours. Failure and Malfunction Report 90-159 was written to document the event. A post trip review of the event was performed in accordance with procedure 1.3.37, "Post Trip Reviews".

BACKGROUND

Prior to the event, steady state operating conditions existed and included the following. The RV water level was +29 inches (narrow range) and the Feedwater System was being controlled in the automatic three element control mode. The RV pressure was 1035 psig and was being controlled via the Electric Pressure Regulator. The Turbine speed was approximately 1800 rpm and Turbine first stage pressure was approximately 730 psig. The Recirculation System pumps were being controlled in the local manual control mode. The Condensate System and Feedwater System pumps were all in service. Except for Bus A6, the APDS was energized by the UAT. The preferred source of offsite power, 345 KV transmission lines 342 and 355, were in service. The 345 KV switchyard air circuit breakers ACB-102 (352-2), ACB-103 (352-3), ACB-104 (352-4), and ACB-105 (352-5) were in service. The 23 KV backup source of offsite power was in service. Thunderstorm activity was reported in the region.

Just prior to the event on May 13, 1990 at 1603 hours, numerous Main Control Room alarms occurred in a short interval of time and included Panel C-3R, "Unit #1 Generator Low Excitation", and "Unit #1 Generator Relay Trip". After the event, the regional power agency (REMVEC) reported that the Pilgrim Station trip occurred at the same time that a 345 KV transmission system line (322) fault occurred. Transmission line 322 is related to the nearby Canal Station and, via transmission line 342, to Pilgrim Station. A static wire (lightning protection) for line 322 fell onto its conductors and caused a fault. The fault resulted in an electrical disturbance that was detected and isolated by protective devices on line 322. The momentary disturbance was sensed by the transmission system prior to the isolation of line 322. The disturbance did not result in a trip of the Canal Station generators but did result in the trip of Pilgrim Station.

CAUSE

The cause for the Pilgrim Station trip was the momentary fault on line 322 (transmitted via line 342) and the absence of the telephone relay time delay ('X' coil) function of the loss-of-field relay (240) at Pilgrim Station. The disturbance actuated the impedance (Z) and directional (D) and voltage (V) units of the relay (240) that is part of the Generator's protective circuitry. The 'X' coil is normally energized and is designed to change state (drop out) 15 cycles (i.e., 0.25 second) after it de-energizes due to a disturbance (Z and D and V). The time delay is provided to prevent a trip of the Generator due to a momentary fault on the 345 KV transmission system. The failure of the 'X' coil resulted in no time delay for relay 240 and consequently, the Pilgrim Station Generator Lockout Relay (286-1) actuated at the time of the momentary fault. The loss-of-field relay (240) was manufactured by Westinghouse Electric Corporation, type KLF-1, style 292B333A10, 125 VDC, 5 Amps, 69 volts, 60 Hertz.

The non safety-related loss-of-field relay (240) is functionally tested in accordance with procedure 3.M.3-39 Attachment 1, "Turbine/Generator Lockout Test and Associated Annunciator Verification" and calibrated in accordance with Attachment 2, "Calibration of Turbine/Generator Relays". The functional test includes a check (operation) of the relay's alarm functions and trip functions. The relay was most recently functionally tested and calibrated with satisfactory results while shutdown on October 26, 1989. The operation of the telephone relay was tested in accordance with the vendor manual (V-0250) that does not include or specify a calibration (timing) of the relay's inherent time delay. The time delay of the relay was built-in and not adjustable, and therefore was not checked as part of the calibration. The relay (240) was installed during original plant construction (c. 1972). A search of the Nuclear Plant Reliability Data System (NPRDS) revealed no other failures of a KLF-1 relay. Therefore, the cause for the open ('X') coil of the telephone relay is believed to be random or age related failure. The cause for the open coil is being investigated and an update to this report will be submitted if the investigation reveals significant new information.

CORRECTIVE ACTION

The unit returned to commercial service on May 15, 1990 at 1030 hours.

The loss-of-field relay (240) was replaced on May 16, 1990 with another relay (KLF-1) having an adjustable time delay. Interim measures taken until the relay was replaced consisted of modifying the relay via a Temporary Modification (TM 90-11) that was supported by a Safety Evaluation (SE 2470). Essentially, the relay's (240) trip function was disabled without affecting the relay's alarm functions.

The relay's (240) calibration sheet has been revised to include a calibration of the new relay's adjustable time delay. The loss-of-field relay (240) is the only one of its type (KLF-1) installed at Pilgrim Station.

An Engineering Service Request (ESR 90-327) has been written to determine if any other relay(s), included in procedure 3.M.3-39 or 3.M.3-40 (Relay House Testing), is equipped with a built-in time delay that may need additional testing.

SAFETY CONSEQUENCES

This event posed no threat to the public health and safety.

The Technical Specification 2.2.B limiting safety system setting for the Main Steam System/Pressure Relief System (PRS) relief valves is 1095 to 1115 psig with a tolerance of +/- 11 psi. The setpoint of the relief valves is 1115 psig. Therefore, the setpoint range of the relief valves including tolerance is 1104 psig to 1126 psig. During the event, the highest RV/Main Steam System pressure that occurred was approximately 1100 psig. The two-stage Target Rock relief valves (RV-203-3A/3B/3C/3D) are installed in Main Steam pipelines 'A' (RV-203-3A and -3D) and 'D' (RV-203-3B and -3C).

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

Review of RV pressure and Main Steam pressure data indicates that the pressure increased rapidly (i.e., 38 psig in 0.5 second) until RV-203-3A (s/n 1040) and RV-203-3B (s/n 1048) actuated. After RV-203-3A and -3B actuated, the pressure continued to increase, but in a less rapid manner, for another 0.5 second until RV-203-3C (s/n 1046) actuated. After RV-203-3C actuated, and with RV-203-3A and -3B still actuated, pressure remained relatively constant for another 1.2 seconds and RV-203-3C then closed. After RV-203-3C closed, and with RV-203-3A and -3B still actuated, pressure decreased (i.e., 30 psig in 2 seconds) and RV-203-3A and -3B then closed. The duration of the pressure transient, from initial to maximum to initial (all relief valves closed), was approximately 5 (five) seconds. Relief valve RV-203-3D (s/n 1025) did not actuate for the pressure relief function. The relief valve (s/n 1025) was most recently tested at an offsite test facility on August 2, 1988 with certified as-left popping pressures of 1114 psig +/- 1 (one) psig. The load rejection with bypass experienced during this event is bounded by the transient analysis described in the Updated Final Safety Analysis Report section 14.4.3, "Generator Load Rejection Without Bypass". The actuation (opening) of some or all of the 4 (four) two-stage relief valves (RV-203-3A/3B/3C/3D) is an expected response to a load rejection with bypass at 100 percent power.

The Technical Specification 2.2.C limiting safety system setting for the Main Steam/PRS safety valves (RV-203-4A and -4B) is 1240 +/- 13 psi. During the event, the highest RV pressure that occurred (1100 psig) was approximately 140 psig less than the safety valves' setpoint (1240 psig).

The HPCIS was manually started in the full flow test mode in accordance with procedure 2.2.21 [HPCIS] section 7.4.2 and in accordance with the guidance of EOP-01. In the full flow test mode, steam from Main Steam pipeline 'D' is supplied to the HPCIS turbine and is exhausted to the Suppression Pool, and water from the Condensate Storage Tanks (CST) is supplied to the HPCIS pump and is returned to the CST via the test return line. The injection function of the HPCIS was not used.

The scram signal was the designed response to the Turbine Control Valves Fast Closure (i.e., opening of PS-37/38/39/40). The closure was the expected designed response to a load rejection with the Turbine first stage pressure at approximately 730 psig, i.e. greater than the scram bypass setpoint (calibrated at 108 psig +/- 3 psig) corresponding to 25 percent of the normal first stage pressure).

The decrease in the RV water level was the expected response to the scram and accompanying shrink in the RV water. The PCIS and RBIS actuations were the expected designed responses to a low RV water level condition, i.e. +12 inches (narrow range).

The Technical Specification 2.1.I limiting safety system setting for actuation of the Core Standby Cooling Systems (CSCS) is -49 inches. During the event, the lowest RV water level that occurred (-10 inches) was approximately 36 inches above the CSCS setpoint (calibrated at approximately -46 inches). In addition, the level (-10 inches) was approximately 117.5 inches above the level that corresponds to the top of the active fuel zone (-127.5 inches).

This report is submitted in accordance with 10 CFR 50.73 (a)(2)(iv) because the RPS was actuated.

SIMILARITY TO PREVIOUS EVENTS

A review was conducted of Pilgrim Station Licensee Event Reports (LERs) submitted since January 1984. The review focused on LERs submitted in accordance with 10 CFR 50.73(a)(2)(iv) that involved the loss-of-field relay, a load rejection or similar scram. The review identified similar events reported in LERs 50-293/85-025-00 and 89-026-00.

For LER 85-025-00, an automatic scram occurred on September 1, 1985 at 0521 hours while at 32 percent reactor power. At the time of the event, the Main Condenser was being backwashed and a live washdown of the 345 KV switchyard insulators was being performed to reduce arcing due to salt from a heavy ocean storm. A 345 KV phase 'B' insulator, located on the Main Transformer side of the switchyard ACB-104, disintegrated and resulted in a load rejection. The cause for the scram was high RV pressure that resulted from the load rejection. The cause for the event was due to the forces of nature (i.e, high winds and salt air). Please note that the event occurred while at 32 percent reactor power. At that power level, the Turbine first stage pressure was approximately 200 psig. An RPS scram signal due to a Turbine Control Valves Fast Closure (i.e., PS-37/38/39/40) or Turbine Stop Valves closure (i.e., less than 90 percent open) would have occurred if the Turbine first stage pressure was greater than approximately 280 psig (i.e., scram bypass setpoint for 45 percent of the normal first stage pressure). The scram bypass setpoint was changed from 280 psig to 108 psig (+/- 3 psig) via a modification (PDC 87-48) during Refueling Outage number 7.

For LER 89-026-00, an automatic RPS scram signal and scram occurred on August 30, 1989 at 1917 hours while at 65 percent reactor power. The cause for the scram signal was high RV pressure (ultimately 1069 psig) that occurred as a result of an automatic Turbine runback. The runback included the automatic adjustment of the Turbine Control Valves and sequential opening of the Turbine Bypass Valves. The runback occurred as a result of the failure of the primary winding of the Main Generator 24 KV phase 'A' potential transformer and a Generator Voltage Balance Relay (260) wiring error that affected the transfer function of the Generator's Voltage Regulator. The wiring error was due to a drawing error. The error was not previously detected because the surveillance test procedure (3.M.3-39) used to functionally test the relay (260), although demonstrating the voltage balance relay function, and alarm functions, did not include a step(s) to identify the auxiliary relay (260X1 or 260X2) that actuates the same alarm (Panel C-3R, "Generator Potential Fuse Blown") during the test(s).

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

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

COMPONENTS	CODES
Relay, Voltage and Power Directional (240)	92
Valve, Relief	RV

SYSTEMS	
Containment Isolation Control System (PCIS,RBIS)	JM
Engineered Safety Features Actuation System (PCIS, RBIS, RPS)	JE
Main Generator Output System	EL
Main Steam System	SB
Main Turbine System	TA
Plant Protection System (PRS)	JC
Reactor Water Cleanup (RWCU) System	CE
Standby Gas Treatment System (SGTS)	BH
Switchyard System (345 KV)	FK