NRC Form 366 (6-89) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
PLANT HATCH, UNIT 1	0 5 0 0 0 3 2 1	1 OF 6

TITLE (4)
BLOWN FUSE RESULTS IN AN UNPLANNED AUTOMATIC ESF ACTUATION AND AN INTERRUPTION OF SHUTDOWN COOLING

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQ NUM	REV	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
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0 4	1 4	9 3	9 3	0 0 4	0 0	0 5	1 4	9 3		0 5 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (11)

OPERATING MODE (9)	5											
POWER LEVEL	0 0 0		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)
			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	
STEVEN B. TIPPS, MANAGER NUCLEAR SAFETY AND COMPLIANCE, HATCH	912	367-7851

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORT TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORT TO NPRDS	

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES(If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (16)

On 4/14/93, at 0910 CDT, Unit 1 was in the Refuel mode and core reload was in progress with 23 fuel bundles loaded in the core. The "A" division of the Residual Heat Removal (RHR) System was out of service for maintenance. The "B" division of RHR was in operation in the Shutdown Cooling (SDC) mode. At that time, while performing a periodic surveillance, a licensed operator found that RHR-SDC flow was at zero. He then checked the system alignment and found that the Low Pressure Coolant Injection valve (valve 1E11-F015B), which also functions as the SDC discharge valve, was closed, isolating SDC from the reactor vessel. The operator opened the valve; however, when the valve reached the full open position, it automatically reclosed. The 1D RHR pump was then secured, fuel movement was stopped, and procedure 34AB-E11-001-1S, "Loss of Shutdown Cooling," was entered. An investigation was subsequently initiated. At 1015 CDT, it was determined that the valve had closed due to a blown fuse. The fuse was replaced and the valve was reopened without incident. By 1043 CDT, the RHR system had been filled and vented and SDC had been returned to service. Reactor coolant temperature did not increase during the event.

The cause of the event was inadvertent grounding of a logic circuit resulting in the circuit fuse actuating. The grounding incident occurred at approximately 0745 CDT during a modification activity not associated with the RHR system.

Corrective actions included replacing the fuse, testing of the SDC pump, returning SDC to service, revising administrative controls, and issuing an operating order.

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor Energy Industry Identification System codes are identified in the text as (EIIS Code XX).

DESCRIPTION OF EVENT

On 4/14/93, at 0910 CDT, Unit 1 was in the Refuel mode and core reload was in progress with 23 fuel bundles loaded in the core. The "A" division of the Residual Heat Removal System (RHR, EIIS Code BO) was out of service for maintenance. The "B" division of RHR was in operation in the Shutdown Cooling (SDC) mode. At that time, while performing periodic surveillance procedure 34GO-OPS-015-1S, "Maintaining Cold Shutdown or Refuel Condition," a licensed operator found RHR-SDC flow at zero. He then checked the system alignment and found the 1D RHR pump running. However, the Low Pressure Coolant Injection valve (1E11-F015B), which also functions as the SDC discharge valve, was closed, isolating SDC from the reactor vessel. The operator opened the valve using the valve control switch. However, when the valve reached the full open position it automatically reclosed, indicating that an automatic closure signal was in effect in the valve control circuit.

The 1D RHR pump was then secured, fuel movement was stopped, and procedure 34AB-E11-001-1S, "Loss of Shutdown Cooling," was entered. Since the "A" division of the RHR system was out of service for maintenance, it could not be aligned to the SDC mode.

Reactor coolant temperature was then monitored every 15 minutes via the Fuel Pool Cooling and Cleanup (FPCC) System (EIIS Code DA). The reactor coolant temperature is normally monitored by using the RHR heat exchanger inlet temperature. However, with SDC out of service, the heat exchanger inlet temperature would not be indicative of reactor coolant temperature. Therefore, reactor coolant temperature was monitored at the suction of the FPCC system which was aligned to the reactor cavity to identify any increasing trends in temperature.

An investigation was initiated regarding the closing of valve 1E11-F015B. At 1015 CDT, it was determined that a blown fuse had caused the valve to close. Based on subsequent review of the RHR system flow recorder chart, it was concluded that the valve had closed, and the fuse had blown, at approximately 0745 CDT. A periodic check of Emergency Core Cooling Systems (ECCS) status had been performed at approximately 0730 CDT at which time the valve had been found to be open.

The fuse was replaced and the valve was reopened without incident. By 1043 CDT, the RHR system had been filled and vented and SDC returned to service. Within 10 minutes following the return to service of SDC, the RHR heat exchanger inlet temperature returned to its pre-event level of 95 degrees Fahrenheit. It was therefore apparent that reactor coolant temperature did not increase during the event. Since only 23 fuel bundles were in the reactor vessel, the heat load was very low and the reactor coolant temperature was not expected to have increased.

TEXT

CAUSE OF EVENT

The cause of the event was inadvertent grounding of a control circuit during implementation of a design modification. During the implementation of the Hardened Vent modification, a panel of a control board had to be removed to gain access to the affected wiring. While the panel was being removed, a valve indicating light terminal on the panel contacted the control board frame, grounding the associated circuit. As a result, the circuit fuse actuated and de-energized the circuit. The circuit was part of the Primary Containment Isolation System (PCIS, EIIS Code JE) logic circuitry and was of a fail-safe design. Consequently, when the circuit de-energized, a PCIS signal was generated resulting in the automatic closure of valve 1E11-F015B.

A licensed operator passing by the panel at the time of the event saw a spark when the terminal was grounded. However, it did not appear that there had been any effect on the plant. Therefore, he did not feel it necessary to investigate further. The contract electricians performing the work also saw the spark but took no action, believing that the operator would take any necessary actions.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required pursuant to 10 CFR 50.73 (a)(2)(iv) in that a blown fuse resulted in an unplanned automatic actuation of an Engineered Safety Feature (ESF). Specifically, the SDC discharge valve (1E11-F015B) also functions as a Primary Containment Isolation System valve. As explained earlier, the fuse actuation resulted in the associated circuit de-energizing. The de-energized circuit simulated a PCIS signal to the valve resulting in it automatically closing.

The purpose of the Primary Containment Isolation System is to provide timely protection against the onset and consequences of events involving the potential release of radioactive materials from the fuel and nuclear system process barriers by isolating appropriate lines which penetrate Primary Containment. Additionally, isolation of the lines acts to conserve reactor water inventory if a breach in the line is causing a loss of reactor coolant. The PCIS logic is of a fail-safe design such that on loss of power to the logic a PCIS isolation signal is generated. In this event, a fuse actuation resulted in the de-energization of a portion of the Group 2 PCIS logic. As a result, PCIS valve 1E11-F015B automatically closed as designed. Other PCIS valves received signals to close; however, with the plant in a refueling outage, most of these valves were already closed prior to the event. The Safety Parameter Display System (SPDS, EIIS Code IQ) was out of service for maintenance at the time of the event and, therefore, could not be used to confirm valve actuations.

The purpose of the Shutdown Cooling mode of the RHR system is to provide adequate cooling to the reactor core while the reactor is shutdown in order to reduce the reactor coolant temperature to and/or maintain it below 212 degrees Fahrenheit. In this event, Shutdown Cooling flow was interrupted for

approximately three hours. At the time of the event, only 23 fuel bundles were loaded in the reactor and it had been thirty days since the core was last critical. Consequently, the decay heat load in the vessel was extremely low and the temporary interruption of SDC flow did not result in an increase in the temperature of the reactor coolant.

Even if the event had occurred under the most limiting plant conditions, the interruption of SDC flow would have been discovered with ample time available for taking the necessary actions to maintain the core covered and adequately cooled. The worst case scenario for a loss of SDC is postulated to occur with the reactor core fully loaded and the vessel head de-tensioned or removed at the earliest possible point after the reactor is shutdown. The earliest point at which this condition could exist is approximately 2.5 days after the reactor is shutdown. At this point in time, the decay heat load would be approximately 37 million BTU/hr. With this heat load, following an interruption of SDC and assuming a conservative initial coolant temperature of 150 degrees Fahrenheit, the bulk reactor coolant temperature would reach 212 degrees Fahrenheit in 1.25 hours at which point boiling would commence. The decay heat load is not sufficient to cause a departure from nucleate boiling and core coverage is sufficient to prevent fuel cladding damage. Once bulk boiling begins, it would take 5 hours for reactor water level to decrease to the top of the active fuel with no makeup. This assumes that prior to the event reactor water is at a level of 195.4 inches above the top of the active fuel. This is also a conservative assumption since the level would actually be at the vessel head flange which is 364.2 inches above the top of the active fuel.

With bulk boiling in progress, it is assumed that the steam generated is vented to the refueling floor. Refueling floor personnel would notice water vapor rising from the reactor vessel cavity prior to the onset of boiling and would notify the Control Room. Additional assurance that the interruption of shutdown cooling would be identified early in the event is described as follows. At 190.4 inches above the top of active fuel, a reactor water low level annunciator will alarm. A reactor scram signal and a group 2 PCIS signal will actuate at 170.7 inches above the top of active fuel. At 123.4 inches above the top of active fuel, the Standby Gas Treatment System will start. At 57.4 inches above the top of active fuel, a group 1 PCIS signal will actuate and the Diesel Generators and Core Spray pumps will start and automatically inject into the vessel thereby terminating the event. These actuations will result in indications and alarms in the Control Room. Even if the operators did not notice the numerous indications and alarms, the condition would be detected during performance of surveillance procedure 34GO-OPS-015-1S long before the reactor water level decreased to the top of the active fuel. The procedure is performed every 4 hours. If the loss of SDC occurred immediately after performance of the surveillance, then the event would progress for 4 hours before the surveillance would be performed again. Bulk boiling would be in progress; however, the reactor water level would be 123.4 inches above the top of the active fuel. At least one division of the Core Spray System (CS, EIIS Code BM) and the Control Rod Drive System (CRD, EIIS Code AA) would be available for adding coolant to the reactor vessel. Either of these systems could be aligned to inject to the reactor vessel, restoring reactor water level and cooling the reactor core.

TEXT

In this event, had the heat load in the reactor vessel at the time the event was discovered warranted immediate mitigating actions, the 1E11-F015B valve could have been manually opened in a minimal amount of time, thus returning SDC to service. The automatic closure signal to the valve would not have prevented the valve from being manually opened.

Additionally, during the event, the 1D RHR pump ran at shutoff head and no flow conditions for approximately 1.5 hours. The pump was secured after it was determined that valve 1E11-F015B could not be maintained open using the remote control switch. When SDC was placed back into service, the pump flow and discharge pressure were checked and found to be acceptable. Pump vibration was checked and found to be acceptable. Also, the 1B RHR pump was available had the 1D pump been damaged during the event.

Based on the above analysis, it is concluded that this event had no adverse impact on nuclear safety. This assessment considers the worst case initial plant conditions and therefore envelopes all other plant conditions.

CORRECTIVE ACTIONS

The fuse was replaced, valve 1E11-F015B was opened, and the SDC mode of the RHR system was placed into service.

A plant-wide directive was issued on this event informing plant personnel of the possible consequences of grounding incidents and the need to aggressively investigate each grounding incident to determine the effects on plant operation.

An operability check and a vibration check were satisfactorily performed on the 1D RHR pump.

Administrative controls will be reviewed and revised as necessary to ensure that SDC flow and reactor coolant temperature are checked at a frequency commensurate with the decay heat load in the reactor vessel.

ADDITIONAL INFORMATION

No systems other than those previously addressed in the report were affected by this event.

TEXT

No events have occurred in the previous two years which resulted in a loss of RHR-SDC capability. Eight events have occurred in the previous two years in which a fuse actuation resulted in the automatic actuation of an Engineered Safety Feature. These events were addressed in the following reports:

50-321/91-16, dated 9/30/91,
50-321/91-21, dated 10/25/91,
50-321/91-23, dated 11/12/91,
50-321/92-16, dated 7/10/92,
50-366/91-10, dated 5/13/91,
50-366/91-11, dated 5/15/91,
50-366/92-02, dated 2/19/92, and
50-366/92-18, dated 10/26/92.

Corrective actions for these events included replacing the actuated fuse, returning the applicable system to service, counseling personnel, training personnel, performing a design review of the fuse application, performing a check of the applicable circuit for faults, and evaluating the usage of different types of jumpers. These corrective actions could not have prevented the event addressed by this report because of the unique circumstances involved in this event. Specifically, in this event, electricians were having to remove a valve light indication panel in order to access wiring. In moving the panel out of its frame, a valve indicating light terminal on the panel contacted the panel frame, grounding the circuit. The corrective actions from the previous events could not have prevented this event.

No failed components resulted from or contributed to this event.