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United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Perry Nuclear Power Plant
Docket No. 50-440
LER 2001-003-00

Ladies and Gentlemen:

Enclosed is Licensee Event Report (LER) 2001-003, Loss of Feedwater Scram and Specified System Actuations including ECCS Injection. There are no regulatory commitments contained in this letter. Any actions discussed in this document represent intended or planned actions, are described for the NRC's information, and are not regulatory commitments.

If you have questions or require additional information, please contact Mr. Gregory A. Dunn, Manager - Regulatory Affairs, at (440) 280-5305.

Very truly yours,

T H Bome

for John K. Wood
Attachment

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III

IE22

NRC FORM 366 (1-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 6-30-2001

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Perry Nuclear Power Plant	05000 440	1 OF 5

TITLE (4)
Loss of Feedwater Scram and Specified System Actuations including ECCS Injection

EVENT DATE (5) MO	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REV NO	REPORT DATE (7) MO	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAME	DOCKET NUMBER
07	11	01	2001	003	00	08	20	2001		05000
										05000

OPERATING MODE (9): 1

POWER LEVEL (10): 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)

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

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Kenneth Russell, Compliance Engineer	1-440-280-5580

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	EC	FU	BUSS	Yes					

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

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

On 7/11/01, at 2223 hours, the plant was operating at 100 percent reactor power when a loss of feedwater occurred resulting in an automatic reactor scram, an automatic trip of the Reactor Recirculation Pumps, an automatic initiation and injection of the High Pressure Core Spray System (HPCS) and the Reactor Core Isolation Cooling System (RCIC) and a Balance of Plant Isolation due to a valid Reactor Pressure Vessel (RPV) water low level signal. Subsequently, at 0400 on 7/12/01, an invalid low RPV water level signal occurred resulting in an invalid automatic initiation of the HPCS pump, HPCS diesel and associated support systems, however no injection occurred. All systems responses were as expected to the valid and to the invalid signals.

The cause of the loss of feedwater event was determined to be a blown fuse that supplies 24-Volt DC electrical power to the Loop B Analog Instrument Panel, which contains instrumentation for the feedwater control system. The cause of the invalid RPV low level actuation was determined to be due to a thermal hydraulic perturbation coincident with decreasing reactor pressure.

The power supply fuse was replaced and a thermographic examination was performed of all fuses located in the Control Room. A thermographic examination of the Control Room fuses was again performed following plant restoration at approximately 60 percent power. Although no additional failures were identified, fuses indicating above normal temperatures were replaced. To address the invalid HPCS initiation, the signal dampening for the HPCS RPV level instruments was increased in order to reduce the instrument vulnerability to short duration pulses.

This event is being reported in accordance with the requirements of 10CFR50.73(a)(2)(iv), an event or condition that resulted in specified system actuations.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Perry Nuclear Power Plant	05000	2001	-- 003	-- 00	2 OF 5

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

I. Introduction

On 7/11/01, at 2223 hours, with the plant operating at 100 percent reactor power, a loss of feedwater occurred resulting in an automatic reactor scram due to a valid Reactor Pressure Vessel (RPV) low water level signal (Level 3, 178 inches above the top of active fuel). In addition, an automatic initiation of the High Pressure Core Spray System (HPCS) [BG], the Reactor Core Isolation Cooling System (RCIC) [BN], an automatic trip of the Reactor Recirculation System Pumps [AD] and Balance of Plant (BOP) and Reactor Water Cleanup System (RWCU) [CE] isolations occurred due to a valid RPV low water level signal (Level 2, 130 inches above the top of active fuel). Proceduralized Operator actions were taken to stabilize reactor level and HPCS was removed from service at 2233 hours with reactor level being maintained with RCIC. The maximum RPV pressure was 1025 psig, which was the pressure prior to the scram. The RPV level was restored by injection by the HPCS and RCIC systems. During the post scram operation/transient, the minimum actual RPV water level experienced was approximately 111 inches above the top of active fuel.

Subsequently, at 0400 on 7/12/01, an invalid RPV level 2 low water level signal occurred resulting in an automatic start of the HPCS pump, HPCS diesel and associated support systems. No injection occurred as a result of the HPCS injection valve being closed due to an actual RPV high level signal (Level 8, 219 inches above the top of active fuel). HPCS was overridden to off in accordance with Off-Normal Instruction "Inadvertent Initiation of ECCS/RCIC (ONI-E12-1)" as a result of the invalid low level signal. The over-ridden start signal was sealed in approximately 1 hour and 40 minutes until the Division 3 initiation signal was reset. This action is directed to prevent inadvertent injection of water into the RPV. No other divisions of Emergency Closed Cooling System (ECCS) were affected.

An NRC notification was made via the Emergency Notification System at 0029 hours on 7/12/01, (ENF No. 38132), in accordance with the requirement of 10CFR50.72 (b)(2)(iv)(A), ECCS Injection, 10CFR50.72 (b)(2)(iv)(B), as an event that resulted in an actuation of the Reactor Protection System (RPS) when the reactor is critical and 10CFR50.72(b)(3)(iv), as specified system actuations, which in this case includes the RPS and HPCS, systems as well as the BOP isolation. This event is being reported in accordance with the requirements of 10CFR50.73(a)(2)(iv), a condition that resulted in multiple specified system actuation's and also satisfies PNPP's Operational Requirements Manual (ORM) section 7.6.2.1, which requires a special report submittal following an ECCS actuation and injection into the reactor coolant system. This was the thirteenth HPCS injection to date. The injection nozzle usage factor is currently less than 0.70.

II. Event Description

At 2223 hours, on 7/11/01, several alarms were activated in the Control Room including Reactor Feedwater Booster Pump D discharge pressure low, Moisture Separator Reheater 1B and 2B drain tank level high, Steam Seal Exhaust system vacuum low, Reactor Feedwater Pump B discharge pressure low, and Heater 4 level low. In addition, the hot surge tank level control subsystem lost power causing the hot surge tank level control valve to close, which started a decrease in hot surge tank level. Also, power was lost to the automatic control subsystem for the Reactor Feed Pump minimum flow recirculation valves causing the valves to open. This resulted in a decrease in Reactor Level, causing the Reactor Feedwater Pumps to increase in speed to make up level, and slightly over-feed the vessel. The decreasing level in the hot surge tank caused a decrease in Reactor Feed Booster Pump discharge pressure, which then caused a decrease in suction and discharge pressure at the Reactor Feedwater Pumps. The Reactor Feedwater Pumps continued to try to compensate by increasing speed until the discharge pressure dropped below reactor pressure and lost feed capability. The subsequent reactor level decrease resulted in a SCRAM on Level 3 at 2227 hours. Reactor level continued to decrease to Level 2 resulting in a valid actuation and injection of the HPCS and the RCIC systems, an automatic trip of the Reactor Recirculation System Pumps, as well as a Balance of Plant isolation at 2229 hours. Proceduralized Operator actions were taken to stabilize reactor level, including removal of HPCS from service at 2233 hours, with reactor level being maintained with RCIC.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Perry Nuclear Power Plant	05000	2001	-- 003	-- 00	3 OF 5

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

The plant shutdown continued, utilizing RCIC and turbine steam bypass valves (BPVs), lowering reactor pressure and temperature, until 0400 hours on 7/12/01. At that time an invalid HPCS and HPCS diesel actuation occurred, including the appropriate support systems. HPCS did not inject since the instrument controlling the HPCS injection valve was demanding the injection valve closed by an actual high RPV level, Level 8. Proceduralized Operator action was taken to over-ride the HPCS pump off. The operators maintained control over the HPCS system, which was available at all times, for manual injection if required. The pump was subsequently returned to standby at 0540 hours. Mode 4, Cold Shutdown, was achieved at 0115 hours on 7/13/01.

III. Cause of Event

The cause of the loss of feedwater event was determined to be a blown fuse in the 24 volt DC circuit that supplies the Loop B Analog Instrument Panel. The 24-Volt DC power supply is supplied by an auctioneered circuit that is fed from one of two 120-volt AC power sources. The loss of 24-Volt DC power caused the failure of the feedwater system controllers and subsequent loss of the feedwater system. Inspection of the fuse body of the blown fuse, showed signs of long term overheating including oxidation at the fuse to fuse clip connection. The fuse is a fast acting fuse that opens immediately when subjected to short circuit current. An internal inspection of the fuse revealed no indication of severe arcing or fusing of the sand crystals as would be expected following a fuse operation under fault current conditions. The fuse element was found open with mild melting at the first indentation closest to the supply end ferrule. Therefore, it is concluded that the fuse operated due to long term heating due to the high resistance connection at the supply side of the fuse clip. The high resistance connection was due to age related oxidation, the fuse had not been previously removed from the fuse clip or replaced.

An investigation team, separate from the SCRAM investigation team, was formed to research the invalid RPV level 2 ECCS actuation. Potential causes that were eliminated include: actual level 2, level notching, electrical transients, trip unit failure, RPV pressure transient (reference leg ringing), level transmitter failure, temperature stratification, EMI/RFI induced (radio interference), air entrainment in reference or variable leg, blockage of sensing lines, digital equipment interference, instrument line leakage and plant changes/design modifications. Additionally, since RCIC was running for this event, it was compared to a February 16, 1987 event in which 6-inch level notches were experienced during RCIC testing at rated pressure. As a result of the 1987 event, reference leg nozzle hoods were installed which eliminated the level notching problem. While this event is different in operating condition and in observed effects from the 1987 event, an inspection of the nozzle hoods will be performed during the next refuel outage to verify that the nozzle hoods are not damage.

The apparent cause of the invalid RPV level 2 ECCS actuation was determined to be due to a thermal hydraulic perturbation, in the "C" reference leg only, similar to that discussed in Perry LER 2001-001 (See Similar Events section). The Rosemount 1153 wide range level transmitters used in the "C" reference leg have a short response time as compared to narrow range instruments which do not respond to the short duration pulses. This allowed the signal perturbation to meet the trip unit's logic for HPCS pump start. The perturbation was caused by either water being introduced into the impulse line between the reactor and the "C" reference leg condensing chamber or that water in the condensing chamber flashed during the vessel depressurization. The duration of the perturbation, approximately 19 milliseconds was too short to cause the injection valve trip unit logic to take control and open the injection valve, whereas the pump start logic seals in causing the pump to start. Since the injection valve logic does not seal-in, the valve position relies on actual level (i.e. Level 8 at the time).

A formal root cause evaluation utilizing external involvement is underway including an evaluation of industry experience for applicability. In addition, General Electric has been contracted to evaluate the methodology and corrective actions from the initial cause evaluation.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Perry Nuclear Power Plant	05000	2001	-- 003 --	00	4 OF 5

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

Safety Analysis

The USAR transient that characterizes this scram event is the "Loss of Feedwater Flow" event described in the Updated Safety Analysis Report (USAR) Section 15.2.7. This transient is categorized as an incident of moderate frequency. The blown fuse is identified in GE report EAS 31-0590 titled "Perry Nuclear Power Plant Critical Failure Point Analysis" as being a single point of failure that will result in a plant SCRAM. The USAR analyzed event commences at full power and normal operating pressure and results in the maximum peak pressure in the bottom of the vessel of 1,087 psig which is below the ASME Code limit of 1,375 psig for the Reactor Coolant Pressure Boundary. Vessel dome pressure does not exceed 1,045 psig. The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment is designed; therefore, these barriers maintain their integrity and function as designed.

The 7/11 event started with the plant at full power and is similar to the USAR event. Vessel dome pressure maximum was 1025 psig, which is below the maximum of 1045 in the analysis. Operator actions taken were consistent with the analysis including shutdown of the HPCS system when required for level control. All major and safety systems were OPERABLE at the time of the event other than the Containment/Drywell Hydrogen Analyzer "A" and the Oscillation Power Range Monitors, neither of which were required to mitigate the events of this transient. Therefore, the evaluation of this event with respect to the USAR is considered bounded by the existing accident analysis.

Although the invalid Division 3 RPV level 2 ECCS level actuation was not assumed within the USAR analysis, the plant systems responded as designed and no significant effects resulted (i.e., there were no equipment failures).

As a consequence to the reactor recirculation pumps tripping off, the RWCU system isolation and HPCS and RCIC systems injecting into the RPV, excessive heatup and cooldown rates were experienced within the RWCU piping exiting the reactor vessel bottom head. The RWCU heatup and cooldown are bounded by the pressure/temperature cycle analysis for scram and emergency events described in step change heatup drawings.

In summary, this event was reviewed and determined to be bounded by the USAR, other Engineering evaluations or procedural controls. Therefore this condition was determined not to be safety significant.

The Plant's staff calculated a Conditional Core Damage Probability (CCDP) value for the loss of feedwater reactor scram event. The Division 3 initiation was not assumed within the analysis, since when the initiation signals occurred, the plant systems responded as designed. The calculated CCDP for the event, assuming feedwater is excluded from recovery, was 3.95E-8. Using NRC guidance of < 1E-6 as a threshold, the event was not considered risk significant.

IV. Similar Events

A review of the recent Perry LERs and Condition Reports did not identify any transients or SCRAMS as a result of blown fuses. Some Condition Reports did identify occurrences involving blown fuses that did not result in plant transients.

Perry LER 2001-001 includes documentation of an invalid actuation of ECCS systems as a result of thermo-hydraulic perturbations in the level reference legs with level transmitters, which pass through transients of very short duration (milliseconds). The perturbations were the result of high water level and significant RPV depressurization. A corrective action was written at that time to determine if increasing the response times of the level transmitters was warranted. This corrective action had not been completed due to the short interval between the two events. If the response time had been adjusted, the wide range instrument response would more closely match that of the narrow range instruments, which were unaffected in this and the LER 2001-001 events.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Perry Nuclear Power Plant	05000	2001	-- 003	-- 00	5 OF 5

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

VI. Corrective Actions

1. The blown fuse and its fuse clip were replaced.

2. Thermography was performed on all fuses located in the Control Room as part of the extent of condition for this investigation. Although no additional failures were identified, fuses and fuse blocks showing evidence of above normal heating were replaced. Thermography was performed on the replaced components and again on all Control Room fuses when the plant was restarted, at approximately 60 percent power, to ensure that no additional circuits were experiencing heating when the fuses were experiencing their full load operating condition. Control Room fuses are being evaluated for addition to the thermography program, with special emphasis on fuses that have been identified as single points of failure that will result in a plant SCRAM.

3. The signal dampening of the HPCS level transmitters was increased to approximately 0.8 seconds to minimize the potential of short duration perturbations in the reference leg causing an actuation of the HPCS system.

4. A physical inspection of the affected level instruments and their reference legs in the vicinity of the instruments racks was performed for leakage. No leakage was identified.

5. An inspection of the RPV water level tap deflector hoods and drywell reference leg components will be conducted no later than the end of Refuel Outage RF0-9.

6. Although air in the instrument lines was ruled out, potential improvement in the fill and vent procedure for the level instruments has been identified and will be incorporated.

Energy Industry Identification System (EIIS) codes are identified in the text as [xx].