NRC Form 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND 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

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Cook Nuclear Plant Unit 1	05000-315	1 of 3

TITLE (4)

Potential Common Mode Failure of Residual Heat Removal Pumps Due to Use of Inaccurate Values

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: Cook Unit 2	DOCKET NUMBER: 05000-316
06	10	1998	1998	– 031 –	01	12	11	1998	FACILITY NAME	DOCKET NUMBER

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

OPERATING MODE (9)	5		20.2201 (b)		20.2203(a)(2)(v)		50.73(a)(2)(i)	50.73(a)(2)(viii)
POWER LEVEL (10)	00		20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)	50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)	73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)	OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)	X	50.73(a)(2)(v)	
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)	Specify in Abstract below or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Mr. Dan Boston, Safety Related Mechanical Engineering Superintendent	616/465-5901, X1863

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

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 June 10, 1998, with both units in Mode 5, it was determined that the Residual Heat Removal (RHR) pump minimum flow (miniflow) controls for both units had a potential design deficiency. Westinghouse Nuclear Safety Advisory Letter 98-002 stated that during a LOCA of a size to allow the RHR/Low Head Safety Injection pumps to inject into the reactor coolant system (RCS) at less than required miniflow, the miniflow valves might cycle repeatedly from open to close until the valves or the valve motors failed. Available miniflow is a combination of accident mitigation flow and bypass flow through the miniflow valves. If the failed valves prevented adequate miniflow, the associated RHR pumps could fail. In accordance with 10CFR50.72(b)(2)(i), "Any event, found while the reactor is shut down, that, had it been found while the reactor was in operation, would have resulted in the nuclear power plant being in an unanalyzed condition that significantly compromises plant safety," and 10CFR50.72(b)(2)(iii), "Any condition that alone could have prevented the fulfillment of the safety function of a system needed to [m]itigate the consequences of an accident," an ENS notification was made at 1140 hours EDT. An interim LER was submitted in accordance with 10CFR50.73(a)(2)(ii) and 10CFR50.73(a)(2)(v).

The primary cause of this event was use of inaccurate miniflow numbers in calculating the valve control set points. It is not known how long or why inaccurate flow was used for the set point calculations. The values had not been verified by testing. Accurate miniflow values have been determined by flow testing. These numbers will be used in calculating set points for new instruments that will be installed. Other systems were evaluated for similar concerns. Several programs have been initiated or improved to identify or prevent similar concerns.

Overall evaluation of this low probability condition determined that the health and safety of the public were not endangered.

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

Conditions Prior to Event
Unit 1 was in Mode 5, Cold Shutdown
Unit 2 was in Mode 5, Cold Shutdown

Description of Event
On June 10, 1998, during a review of Westinghouse Nuclear Safety Advisory Letter (NSAL) 98-002, engineers determined that the Residual Heat Removal (RHR) pump minimum flow (miniflow) controls for both units had a potential design deficiency. The NSAL stated that during a LOCA of a size to allow the RHR/Low Head Safety Injection pumps to inject into the reactor coolant system (RCS) at less than required miniflow, the miniflow valves might cycle repeatedly from open to close until the valves or the valve motors failed. Available miniflow is a combination of accident mitigation flow and bypass flow through the miniflow valves. If the failed valves prevented adequate miniflow, the associated RHR pumps could fail.

An earlier condition report (CR) investigation had determined that the flow measurement instrumentation that controls the RHR pump miniflow valves would need to be replaced by instruments with different design characteristics. This work had begun in October 1996 when a CR was written because the Unit 2 East RHR Pump failed a post maintenance test. The CR investigation determined that the flow measurement instrumentation that controls the RHR pump miniflow valves would need to be replaced by instruments with different design characteristics. New instruments were being procured when NSAL-98-002 was received. Review of NSAL-98-002 prompted instrumentation and controls (I&C) engineers working on the instrument replacement to focus on potential valve cycling problems. To prevent valve cycling, it was necessary to have an accurate value for the flow through the miniflow line to properly set the open and close setpoints. Ultrasonic flow measurement equipment was used to determine that actual miniflow was approximately 508 gallons per minute (gpm) for Unit 1 and 535 gpm for Unit 2. Once miniflow was known, it was possible to review historical miniflow instrument calibration data and determine if cycling could have occurred in the past. The review showed that the open and close set points, with flow instrumentation calibrated to the historical standards, did not have enough separation to prevent cycling, given the accident scenario presented in NSAL-98-002. The typical open setpoint was about 455 gpm and the typical close setpoint was about 939 gpm. Calibration records showed that with instrument drift and uncertainty, there were periods when the set points did have enough separation to prevent cycling.

Cause of Event
The primary cause of this event was use of inaccurate miniflow numbers in calculating the valve control set points. Determination of the proper control set points depends on accurate knowledge of full flow in the miniflow lines. The actual flows are approximately 508 gpm for Unit 1 and approximately 535 gpm for Unit 2, as determined by recent ultrasonic flow meter testing. The value used historically, which had not been verified by testing, was approximately 463 gpm. It is not known how long or why inaccurate flow was used for the set point calculations.

Analysis of Event
This condition was determined to be reportable in accordance with 10CFR50.72(b)(2)(i), as a condition which was found while the reactor is shut down, which if it had been found while the reactor was operating, would have resulted in the nuclear power plant being in an unanalyzed condition and with 10CFR50.72(b)(2)(iii), as a condition that alone could have prevented the fulfillment of the safety function of a system needed to mtigate the consequences of an accident. An ENS notification was made on June 10, 1998, at 1140 hours EDT on June 10, 1998. An interim LER was submitted on July 10, 1998, in accordance with 10CFR50.73(a)(2)(ii) and 10CFR50.73(a)(2)(v). This LER is submitted as an update.

The safety function of the RHR system is to provide low head emergency core cooling flow during a LOCA. RHR injection may be precluded during a small break LOCA. In such a situation, the miniflow controls play an important role in pump protection by regulating flow through the miniflow lines.

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Analysis of Event (continued)

The event had a low probability of occurrence because multiple conditions would have had to occur in specific sequences to have caused a common-mode failure of the RHR pumps. Cycling could have occurred only if the flow through an RHR train was within a narrow range of values. The approximate flow through a train would have had to have been between 390 and 470 gpm. This flow would have only occurred while the RHR pumps were discharging to the suctions of the safety injection (SI) and centrifugal charging pumps while the RCS was at a relatively high pressure. Even if the flow would have been within the range, the systems were not always susceptible to cycling. Flow instrument drift caused the actual differential between the open and close set points to vary. If cycling had occurred, the valve would have had to have failed closed to deprive the pumps of miniflow. Even with the valve fully closed, flow would have been at least 390 gpm. Westinghouse had informed AEP that the miniflow requirement for similar pumps at another nuclear power plant was approximately 330 gpm. Although this cannot be directly applied to D. C. Cook, it is reasonable to believe that an RHR pump can survive at flows less than the 500 gpm given in the vendor manual.

Finally, there is no reason to believe that cycling would have caused both valves to fail at the same time. The failure of one valve and pump would have allowed RCS pressure to decrease as input flow was reduced. This would have caused the other pump's flow to increase beyond the range where cycling would have occurred. The flow into the RCS would have had to decrease back to the cycling range before the other valve and pump could have failed. The time between the two events would have given the operators time to take corrective actions. The combined effect of the above conditions was to reduce the probability of a common-mode failure.

Overall evaluation of the condition determined that the health and safety of the public were not endangered.

Corrective Actions

Accurate miniflow numbers have been determined by flow testing. These numbers will be used in calculating set points for new flow control instruments that will be installed. The calculation, ECP-12-I3-01, has not yet been completed, however the methodology is complete and is not expected to change. The calculation will serve as the record for how and why the set points were established.

During a review of other systems, engineers determined that the centrifugal charging pumps and SI pumps might be subject to similar conditions. Evaluation of the SI pumps determined that they were not susceptible to the same failure mechanism because there is no automatic control scheme. Evaluation of the charging pumps determined that the associated miniflow system was not as tightly coupled as the RHR miniflow control system. The potential for cycling had been considered during preparation of Calculation ENSM 971023CV, which had established the charging pump miniflow control set points. The calculation basis will be maintained through the new calculation procedure, 800000-LTG-5400.02 "Calculations".

Control and documentation of changes to plant instrument set points have been improved and are controlled by procedure PMP 6065.ISP.001, "Plant Instrument Set Point Control Program."

The operating experience review program, system readiness reviews, restart walkdowns, the calculation verification program, and the set point control and instrumentation uncertainty review, will provide additional assurance that issues similar to the miniflow valve cycling issue are corrected or prevented.

Previous Similar Events

None