NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Grand Gulf Nuclear Station - Unit 1	0 5 0 0 0 4 1 6	1 OF 0 4

TITLE (4) Emergency Core Cooling Systems Delta Pressure Instrumentation Not Calibrated In Accordance with Technical Specifications

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)
0 1	0 9	8 8	8 8	— 0 0 3 —	0 1	0 3	3 1	8 8	NA	0 5 0 0 0
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THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

OPERATING MODE (9)	1				
POWER LEVEL (10)	0 8 4				
		[ ] 20.402(b)	[ ] 20.405(c)	[ ] 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)	[X] 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	
Paul M. Different/Plant Licensing Engineer	6 0 1	4 3 7 - 2 1 6 7

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

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 January 6, 1988 Grand Gulf Nuclear Station Unit 1 resumed power operations following the second refueling outage. On January 9, 1988 the line break instrumentation for Residual Heat Removal (RHR) "A" high differential pressure (dp) alarmed and sealed in. An investigation of the instrumentation revealed that it was working as designed and sensing the actual dp between the RHR "A" and Low Pressure Core Spray (LPCS) injection lines. It was concluded that no actual line break had occurred in either RHR "A" or LPCS piping in the reactor downcomer annulus. The cause of the alarm was determined to be a change in the normal indicated dp between these two Emergency Core Cooling Systems injection lines. The instrument had not been calibrated to actual pressures experienced at 100 percent power following the second refueling outage.

The LPCS, RHR "A" and High Pressure Core Spray (HPCS) line break instrument setpoints have been recalibrated based on the results of an engineering analysis of actual differential pressures. The RHR "B" and "C" setpoints did not require recalibration. These setpoints are intended to detect a fully severed and separated line break at greater than 80 percent power and 90 percent core flow.

Alarm response procedures have been revised to show that at lower power and core flow conditions, the annunciators provide no useful function. The Integrated Operating Instruction for power operations also has been revised to verify the annunciators have cleared after reaching 90 percent core flow.

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NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

A. REPORTABLE OCCURRENCE

On January 9, 1988 the line break instrumentation for Residual Heat Removal (RHR) "A" high differential pressure (dp) alarmed and sealed in and the plant entered the associated Limiting Condition for Operation action statement. Upon further investigation it was determined that the RHR "A" and Low Pressure Core Spray (LPCS) line break instrumentation was not calibrated to normal indicated pressures experienced at 100 percent power conditions. This condition is reportable pursuant to 10CFR50.73(a)(2)(i)(B).

B. INITIAL CONDITION

On January 9, 1988 when the alarm occurred, the plant was in Operational Condition 1 at 84 percent of rated thermal power.

C. DESCRIPTION OF OCCURRENCE

On January 6, 1988 Grand Gulf Nuclear Station Unit 1 resumed power operations following the second refueling outage. On January 9, 1988 at 0630 the line break instrumentation for RHR "A" high dp alarmed and sealed in. The instrumentation involved is transmitter 1E31-N080A (EIIS code GG-1IJ-PDT-N080A) and associated trip unit 1E31-N680E (EIIS code GG-1IJ-PDIS-N680E). The indicated high dp was 1.7 psid at 84 percent power. As power was increased to 100 percent at 104 percent core flow, the indicated dp increased to 2.1 psid. An unrelated scram caused by main transformer failure occurred on January 10, 1988 and the dp dropped to 1.7 psid five minutes after the scram. The dp gradually decreased over the next day and a half to 0.65 psid as reactor pressure decreased to 140 psig.

The RHR "A"/LPCS line break leak detection circuit monitors dp between the two injection lines. Normal dp is expected to be small, since both lines penetrate the vessel and the core shroud. Should one of the headers break between the vessel wall and the shroud, the dp will change since the line that is broken will be exposed to the lower pressure existing in the vessel downcomer region. An alarm would alert control room personnel to this abnormal condition.

Prior to the scram an investigation was performed to determine the cause and validity of the alarm and to determine if the instrumentation was working as designed. This investigation showed the instrumentation was working as designed and accurately measuring dp.

It was concluded from the magnitude of the dp drop after the scram that no RHR "A" line break had occurred. The actual observed drop was 0.4 psid. Had a line break occurred, the dp would have changed significantly more than 0.4 psid. After a scram, reactor recirculation pumps shift to slow speed greatly reducing core flow and steam production which causes the dp across the shroud to decrease to near zero. However, the measured header dp only dropped 0.4 psid indicative of a change in core flow and density effects on the measured dp rather than a break in the header.

D. APPARENT CAUSE

This condition resulted from the use of a calculated dp value in the plant surveillance procedure instead of obtaining actual normal indicated dp as required by the plant Technical Specification.

During startup from the second refueling outage, it was found that the RHR "A"/LPCS line break detection trip unit setpoint was set at a value selected early in the original plant startup program. The selected setpoint was specified in a surveillance procedure. But due to a lack of supporting documentation, the basis for this setpoint could not be determined.

The normal value for the line break dp changed during startup from the second refueling outage. This change pointed out the need to check the setpoint after each refueling outage. The need was not previously recognized because the normal dp value was not expected to change. Therefore, plant procedures were not set up to check for the change in the normal dp.

E. SUPPLEMENTAL CORRECTIVE ACTION

Extensive testing and troubleshooting was performed by Instrumentation and Control (I&C) personnel. These efforts demonstrated that the line break instrumentation is performing its design function for the specified parameter. Additional data was needed to determine if the observed dp increase is temporary or if a new normal indicated dp value is required. The data required included the pressure indicated by each of the dp instrument sensing lines with the corresponding reactor power and core flow rate. The data was recorded in hourly intervals during startup and reactor power increases to nominal full power following the January 10, 1988 scram.

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

Engineering evaluated the dp data from two consecutive startups for the Emergency Core Cooling System (ECCS) injection and core spray line break instrument channels. The data shows that the dp range is 1.5 to 2.1 psid for the LPCS and RHR "A" channels for greater than 80 percent power and 90 percent core flow. The RHR "B" and "C" channels are constant at 0.08 psid over this same power and flow range, while the High Pressure Core Spray (HPCS) channel dp is a constant - 6.2 psid for these conditions.

Based on these new indicated normal dp's, the line break instrument setpoint for HPCS, LPCS, and RHR "A" have been recalibrated. The existing setpoint for RHR "B" and "C" is acceptable as is with no recalibration necessary.

Additionally a review of Technical Specification surveillance requirements was performed to ensure there were no other incidences where normal indicated values were required but not used. This review did not discover any other incidences.

Also, after each future refueling outage the normal indicated dp will be verified at 100 percent power conditions and the setpoints re-adjusted if necessary. Applicable plant procedures will be changed to reflect implementation of this action prior to the next refueling outage presently scheduled for February 20, 1989.

Alarm response procedures have been revised to show that at lower power and core flow conditions, the annunciators provide no useful function. The Integrated Operating Instruction for power operations has been revised to verify the annunciators have cleared after reaching 90 percent core flow.

F. SAFETY ASSESSMENT

There are no safety consequences as it was demonstrated that no piping failures in either RHR "A" or LPCS had occurred. In-service inspections were performed during the recent refueling outage for the LPCS piping inside the reactor vessel. No breaks or problems with the piping integrity were found. The line break instrumentation is not safety related and has only an alarm function which does not actuate any safety related equipment. Investigation concluded that the line break instrumentation is working as designed. Final analysis by engineering has determined that LPCS, RHR "A" and HPCS setpoints required recalibration. RHR "B" and "C" setpoints did not require recalibration.

SYSTEM ENERGY RESOURCES, INC.

OLIVER D. KINGSLEY JR
Vice President
Nuclear Operations

March 31, 1988

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Document Control Desk

Gentlemen:

SUBJECT: Grand Gulf Nuclear Station
Unit 1
Docket No. 50-416
License No. NPF-29
Emergency Core Cooling Systems
Delta Pressure Instrumentation
Not Calibrated In Accordance
With Technical Specifications
LER 88-003-01
AECM-88/0070

Attached is Licensee Event Report (LER) 88-003-01 which is a final report.

Yours truly,

John G. Cesare
FOR
ODKINGSLEY, JR.

ODK:bms
Attachment

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