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October 12, 1995

U.S. Nuclear Regulatory Commiss
Mail Station P1-137
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Grand Gulf Nuclear Station
Docket No. 50-416
License No. NPF-29
Inadvertent Reactor SCRAM on Low Reactor Water
Level Due to Partial Loss of Feedwater
(LER 95-011)

GNRO-95/00114

Gentlemen:

Attached is Licensee Event Report (LER) 95-011 which is a final report.

Yours truly,

DSPace
for CRH

CRH/JEO/
attachment: LER 95-011
cc: Mr. J. E. Tedrow (w/a)
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JE22

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95
LICENSEE EVENT REPORT (LER)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), 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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Grand Gulf Nuclear Station, Unit 1	**05000-416**	**01 of 04**

TITLE (4)
Inadvertent Reactor SCRAM on Low Reactor Water Level Due to Partial Loss of Feedwater

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	DOCKET NUMBER
									N/A	05000
									FACILITY NAME	DOCKET NUMBER
09	17	95	95	011	00	10	12	95	N/A	05000

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

OPERATING MODE (9)	1		20.402(b)		20.405(c)	X	50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	100		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)	X	OTHER
			20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in abstract below and in text, NRC Form 366A)
			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)		Special Report

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
James Owens / Licensing Specialist	**601-437-6483**

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
B	SJ	V	C665	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 15 single-spaced typewritten lines) **(16)**

On September 17, 1995, at approximately 2200 hours, GGNS Operations personnel were conducting the Reactor Feed Pump Turbine Emergency Governor Exercise Monthly Test of the "B" Reactor Feed Pump Turbine. During the course of the test, a trip of the "B" Reactor Feed Pump Turbine occurred, followed by a Reactor Protection System actuation on low reactor water level and a reactor SCRAM. This Reactor Protection System actuation is reportable pursuant to 10CFR50.73(a)(2)(iv).

The low reactor water level Reactor Protection System actuation occurred due to the trip of the "B" Reactor Feed Pump Turbine and failure of the associated pump's discharge check valve to close quickly enough to prevent reverse flow from the "A" Feed Pump back through the "B" pump. Investigation revealed that wear of the valve disc stop allowed the valve disc to open at an angle beyond manufacturer's specifications. The "A" and "B" check valves were corrected by weld build-up of the disc stops.

This event resulted in an Emergency Core Cooling System, High Pressure Core Spray discharge into the Reactor Coolant System as the result of a valid Engineered Safety Feature signal and is being reported as a Special Report in accordance with GGNS Technical Requirement Manual section 7.7.2.1.

NRC FORM 366A (5-92) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95
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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

A. Reportable Occurrence

On September 17, 1995, at approximately 2200 hours, GGNS Operations personnel were conducting the Reactor Feed Pump Turbine (RFPT) [SJ] Emergency Governor [JK] Exercise Monthly Test (04-1-03-N21-5) of the "B" RFPT. During the course of the test, a trip of the "B" RFPT occurred, followed by a Reactor Protection System (RPS) [JC] actuation and reactor SCRAM. This RPS actuation is reportable pursuant to 10CFR50.73(a)(2)(iv).

This event resulted in an Emergency Core Cooling System (High Pressure Core Spray (HPCS) [BG]) initiation as the result of a valid Engineered Safety Feature (ESF) [JE] signal and is reportable in accordance with GGNS Technical Requirement Manual section 7.7.2.1

B. Initial Conditions

At the time of the event the reactor was in OPERATIONAL CONDITION 1 with reactor power of approximately 100 percent. Reactor temperature was approximately 525 F and reactor level approximately 37 inches.

C. Description of Occurrence

On September 17, 1995, at approximately 2200 hours, GGNS experienced an automatic RPS reactor SCRAM while performing the RFPT Emergency Governor Exercise Monthly Test (04-1-03-N21-5) of the "B" RFPT. During this test the overspeed trip is bypassed, however, due to a malfunctioning control fluid pressure regulating valve, the "B" RFPT tripped on an erroneous active/inactive thrust bearing wear signal.

GGNS is designed to handle the trip of a single feed pump. However, the discharge check valve for the "B" pump did not close following the RFPT trip, allowing the "A" pump to short cycle (reverse flow) through the "B" pump. This reverse flow condition diverted the majority of the feedflow from the reactor vessel, thereby preventing the "A" Reactor Feed Pump alone from maintaining reactor vessel water level. As a result, an automatic reactor SCRAM occurred on low reactor water level (Level 3, +11.4 inches).

Vessel level decreased to approximately -40 inches wide range. The Reactor Core Isolation Cooling (RCIC) [BN] system did not initiate and no Containment Isolation (CI) [JM] occurred. Channels "C" and "R" of HPCS initiation logic was made up on low level, initiating HPCS flow into the reactor vessel. The HPCS system in conjunction with "A" Reactor Feed Pump flow not short cycled back through the "B" pump, quickly restored reactor vessel level. The reactor level was stabilized and the HPCS system was secured.

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

An investigation was conducted to determine appropriate system responses for the conditions experienced. A check of the logic and review of the post trip data revealed that the Level 2 logic for RCIC and CI had not been made up. Indications are that the HPCS initiated and quickly turned level upward before the combination of RCIC or CI channels necessary for initiation could be activated. Therefore RCIC and CI systems functioned properly. All control rods inserted fully into the reactor core and there were no Safety Relief Valve (SRV) actuations.

D. Apparent Cause

The cause of the low level RPS SCRAM was a partial loss of feedwater. Contributing causes for this event were the "B" RFPT trip and the open discharge check valve.

E. Corrective Actions

Immediate Corrective Actions:

1. An investigation was conducted to determine why the "B" Reactor Feed Pump discharge check valve did not close. Material Non-Conformance Reports (MNCR 0119-95) were initiated to address the check valve problem.

2. An investigation was conducted to determine whether all systems responded properly for the conditions experienced.

3. An investigation was conducted to determine why the "B" RFPT tripped. MNCR 0254-95 was initiated.

Conclusions:

1. Investigation revealed that the angle of travel for the disc from full closed to full open was 57 degrees. The manufacture's specification for disc travel is 51 to 53 degrees maximum. It was also discovered that the valve disc stop was worn. The wear allowed the disc stop to contact the seat ring at the edge, which increased the potential for binding. This condition could account for the excessive valve disc travel and excessive friction which contributed to slower disc closure and reverse flow. The valve disc stop was repaired and disc travel adjusted to manufacture's specifications by weld buildup. Additionally, a spring loaded plunger assembly used to assist valve disc closure when installed in vertical applications was found missing during RF06 inspections. The vendor stated that the plunger was not necessary since GGNS application of the valve was horizontal rather than vertical. Therefore the plunger was not reinstalled. However, following this event, the plungers were reinstalled in both "A" and "B" valves to provide added assist in initial closure of the valves.

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2. During this event, flow from the "A" Reactor Feed Pump was short cycled (diverted from the reactor). A small amount of flow continued to feed the vessel. Vessel level decreased to approximately -40 inches. The HPCS system initiated and quickly turned level in the positive direction. This prevented the right combination of RCIC, or CI channels from making up on low level. Therefore, all systems responded properly.

3. The initial trip of the "B" RFPT was due to a loose nut on a control fluid pressure regulating valve. The nut maintained a seal between the diaphragm and valve stem. As the nut loosened, regulated pressure was increased until the "B" RFPT tripped on an active/inactive thrust bearing wear signal. The valve was replaced with a new valve.

Long Term Corrective Actions:

The Design Engineering Department will further evaluate the problems experienced with these feedwater check valves and determine whether enhancements can be developed to improve long term performance.

F. Safety Assessment

The reactor SCRAM that occurred on September 17, 1995, did not pose an increased risk to the health and safety of the general public. Upon receiving the RPS signal, all control rods inserted fully. There were no manual or automatic SRV actuations. The HPCS system initiated and in conjunction with "A" Reactor Feed Pump flow not short-cycled back through the "B" pump, quickly restored reactor level to normal range. There were no other ECCS actuation.

The lowest recorded vessel water level for this event was approximately -40 inches on wide range indication (B21R623A), which is 127 inches above the top of active fuel. The RCIC system and CI system logic which also initiate at Level 2, did not actuate. It was concluded that the HPCS system initiated and quickly turned level upward before the setpoint for actuation of RCIC and CI to be reached. A check of RCIC and CI initiation logic indicated that all channel were operable.

G. Additional Information

HPCS injected at a flow rate of approximately 2100 gpm. The temperature of the injection source water was approximately 114 degrees F. The vessel was at 930 psig at the time of the injection. This is the fourteenth (#14) cycle of the HPCS system experienced at GGNS at power. The current value of the nozzle usage factor is still within 0.70. Report of the ECCS injection is being submitted as part of this Licensee Event Report in accordance with the Special Reporting requirements of GGNS Technical Requirements Manual section 7.7.2.1.

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