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December 23, 1993

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

Attention: Document Control Desk

SUBJECT: Grand Gulf Nuclear Station
Unit 1
Docket No. 50-416
License No. NPF-29
Reactor Core Isolation Cooling System Not Operable As Required By Technical Specification
LER 93-017-00

GNRO-93/00166

Gentlemen:

Attached is Licensee Event Report (LER) 93-017 which is a final report.

Yours truly,

DS Pace

CRH/MJM/LFD/BAB
attachment
cc: Mr. R. H. Bernhard(w/a)
Mr. H. W. Keiser(w/a)
Mr. R. B. McGehee (w/a)
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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

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TITLE (4)
Reactor Core Isolation Cooling System Not Operable As Required By 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 NAME	DOCKET NUMBER
									N/A	05000
11	26	93	93	017	00	12	23	93	N/A	05000

OPERATING MODE (9)	2	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § (Check one or more (11))							
			20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	<001		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
			20.405(a)(1)(iii)	X	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)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Riley Ruffin / Licensing Specialist	601-437-2167

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 15 single-spaced typewritten lines) (16)

While commencing unit startup following the sixth refueling outage, iron concentration exceeded EPRI guideline limits for feedwater. Operations personnel were performing appropriate portions of the reactor core isolation cooling (RCIC) surveillance when reactor pressure exceeded the limit specified by GGNS Technical Specification (TS) 3.7.3 with RCIC inoperable. This resulted in noncompliance with TS 3.0.4.

The licensed operator in charge of unit startup had not ordered that control rod movements be ceased soon enough. A contributing factor was that the digital pressure gauge used for monitoring the reactor pressure actually indicates steam pressure for the high pressure turbine first stage. This gauge indicates pressure at about five to eight psig less than steam dome pressure. Other pressure indications on the same control room panel are more difficult to monitor.

Personnel involved in the event were counseled. The need for increased awareness during unit startup was re-emphasized to operators along with discussion that a fast unit startup should not be a priority. The startup procedure did not specify actions to be taken when feedwater chemistry is out of specification and contained inadequate instructions for suspension of control rod movements so that compliance with TS 3.0.4 is maintained. An additional enhancement being considered is that the condensate and feedwater systems be put in their water chemistry cleanup cycles earlier as part of the unit startup evolution in order to preclude the limitations of out-of-spec feedwater.

Health and safety of the general public were not compromised by this event.

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

A. Reportable Occurrence

The limiting condition for operation specified by GGNS Technical Specification (TS) 3.7.3 requires that the reactor core isolation cooling (RCIC) system [BN] to be operable in Operating Conditions 1, 2, and 3 with reactor steam dome pressure greater than 135 psig. Contrary to this requirement, reactor pressure exceeded the specified limit during startup prior to RCIC being operable. This condition resulted in the plant entering an operating condition outside of its technical specifications which is contrary to TS 3.0.4. This event is being reported pursuant to 10 CFR 50.73(a)(2)(i)(B).

B. Initial Condition

The reactor was in Operational Condition 2 with reactor water at approximately 359 degrees F and 138 psig. Plant personnel had been withdrawing reactor control rods as part of unit startup activities following the sixth refueling outage. Main steam line drain valves were open and the reactor vessel was heating up. The swelling reactor water level had been controlled during reactor heatup by rejecting water to the main condenser via the reactor water cleanup (RWCU) system [CE] blowdown. Plant chemistry personnel had recently determined that iron concentration in the reactor feedwater [SJ] exceeded the EPRI guideline limit.

C. Description of Occurrence

While plant operations personnel were performing unit startup activities following the sixth refueling outage, a significant amount of iron and oxygen became entrained in the condensate and feedwater. Plant chemistry personnel had recommended that feedwater not be used to provide makeup to the reactor vessel due to the feedwater being out of specification. Iron concentration exceeded the EPRI guideline limit for feedwater.

On November 26, 1993, reactor control rods were being withdrawn to increase reactor pressure in accordance with the startup procedure, Integrated Operating Instruction (IOI) 03-1-01-1. Upon reaching 60 psig vessel pressure, the IOI directs operations personnel to place RCIC in standby (i.e., ready-for-service) status. The IOI also specifies that RCIC is to be operable prior to exceeding 135 psig reactor steam dome pressure. In addition, the procedure also notes that RCIC isolation valves can not be opened prior to exceeding their low pressure isolation setpoint.

The IOI requirement that RCIC be verified to be in standby is satisfied by performing applicable portions of the RCIC monthly functional test per procedure 06-OP-1E51-M-0001. This action was initiated by operations supervisory personnel in accordance with the IOI. Operations personnel were performing appropriate portions of the RCIC monthly functional test when vessel pressure was observed to approach 125 psig. Operations supervisory personnel then ordered that control rod

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movements be ceased in order to slow the reactor pressure increase until the RCIC test had been completed satisfactorily. Control rod movement was suspended at approximately 125 psig as indicated on panel 1H13P680. Control room panel 1H13P680 is the primary panel which contains controls for the reactor and some of its auxiliary systems.

Reactor pressure continued to increase. Reactor steam dome pressure was determined to have exceeded 135 psig when reactor engineering personnel checked the plant computer. The computer reading indicated that reactor steam dome pressure was 138 psig. Operations staff determined that the computer pressure indication was more precise than the panel gauge and therefore, the RCIC system did not comply with GGNS technical specifications. TS 3.7.3 requires that the RCIC system be operable in Operational Conditions 1, 2, or 3 with the reactor steam dome pressure greater than 135 psig. This event resulted in the plant entering an operating condition outside of its technical specifications which is contrary to TS 3.0.4.

The RCIC monthly functional test was in progress at the time that steam dome pressure was observed to have exceeded 135 psig. The RCIC functional test was completed less than ten minutes after pressure had exceeded 135 psig.

D. Apparent Cause

The licensed operator in charge of unit startup had not ordered that control rod movements be ceased soon enough to prevent noncompliance with TS 3.0.4. The RCIC surveillance had commenced in accordance with the startup procedure, but was not completed satisfactorily prior to the technical specification being applicable.

A contributing factor was that the pressure gauge used for monitoring the reactor pressure did not accurately depict reactor steam dome pressure. The IOI simply specified that reactor pressure was not to exceed 135 psig prior to RCIC being operable; the procedure did not specify reactor steam dome pressure. The digital pressure gauge on panel 1H13P680 that was used to monitor reactor pressure actually indicates steam pressure for the high pressure turbine first stage. This gauge was preferred because it features a digital display. This gauge indicates pressure at about five to eight psig less than steam dome pressure since it senses pressure downstream of the open main steam line drains. Other pressure indications on the same panel include the narrow range and wide range reactor pressure recorders, both which are more difficult to monitor.

Another contributing factor was that the IOI did not contain adequate instruction for suspension of control rod movements and control of reactor pressure so that compliance with TS 3.0.4 is maintained.

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An additional detail of this event was that adequate controls were not utilized when feedwater chemistry was determined to be out of specification. During this phase of the unit startup, operations personnel decided to refrain from supplying the reactor vessel with the out-of-spec feedwater. Operations personnel attempted to control vessel pressure and water level without using feedwater. Operators were concerned about opening main steam bypass valves to control reactor pressure since that action would cause vessel water level to decrease. Adequate controls for the out-of-spec feedwater should have been incorporated into the startup instructions.

E. Corrective Actions

Personnel involved in the event were counseled. The need for increased awareness during unit startup after a refueling outage was re-emphasized to operators along with discussion that a fast unit startup should not be a priority. This event will be discussed with all licensed operators emphasizing control of reactor pressure, level, and power during startup.

The IOI will be changed to specify that control rod movement shall be suspended early enough to comply with the limiting condition for operation specified by TS 3.7.3. In addition, operations and chemistry personnel will discuss implementation techniques for the EPRI guidelines. The Off-Normal Emergency Procedure (ONEP) will be revised to reflect EPRI guidelines for applicable operating conditions. Enhancements to other applicable procedures will be incorporated as warranted to preclude a recurrence. Furthermore, this event was reported under the plant's quality deficiency program via QDR 93-0287 and QDR 93-0297 to review the event and determine corrective actions.

An additional enhancement being considered for future refueling outages is that the condensate and feedwater systems be put in their water chemistry cleanup cycles earlier as part of the unit startup evolution in order to preclude the limitations of out-of-spec condensate and feedwater.

F. Safety Assessment

RCIC was in standby condition less than ten minutes after the noncompliance with TS 3.0.4 occurred. This incident did not impair any other safety system function. All ECCS systems were available to perform their safety function. Health and safety of the public were not compromised by this condition.

G. Additional Information

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