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April 15, 1994

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
Pressure Boundary Leakage Due to Thermowell Failure
LER 93-014-01

GNRO-94/00050

Gentlemen:

Attached is Licensee Event Report (LER) 93-014-01 which is a final report.

Yours truly,

CRH/RR/ for
attachment
cc: Mr. R. H. Bernhard(w/a)
Mr. H. W. Keiser(w/a)
Mr. R. B. McGehee (w/a)
Mr. N. S. Reynolds (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)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Grand Gulf Nuclear Station	05000-416	01 of 03

TITLE (4)
Update to Pressure Boundary Leakage Due to Thermowell Failure

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	02	93	93	014	01	04	15	94	N/A	05000

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

OPERATING MODE (9)	5								
POWER LEVEL (10)	000								
			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
			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)	X	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
X	AD	TW	R369	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 October 27, 1993, plant personnel identified a leak from the thermowell associated with the temperature element for the suction of the 'B' reactor recirculation water pump. Evaluations determined the thermowell had a through-wall leak. The thermowell was replaced and retested satisfactorily. The remaining reactor recirculation thermowells were inspected and no evidence of leakage was identified. Based on subsequent reviews, it was concluded that the most probable cause of failure is flow-induced high cycle low stress fatigue. The current design will be evaluated to determine if redesign is necessary. The failed thermowell did not place the plant in an unanalyzed condition. The leakage that would be experienced in the event of complete failure of the thermowell would have been well within the bounds of the GGNS accident analyses. The failure did not degrade any systems needed for emergency core cooling. Therefore the health and safety of the public were not compromised.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)	PAGE (3)
Grand Gulf Nuclear Station	05000-416	93-014-01	2 OF 03

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

A. Reportable Occurrence

Plant personnel discovered a leak at a thermowell (TW) on the suction side of the Reactor Recirculation Pump [AD]. This component is a part of the reactor pressure boundary and its failure represents a degradation in the integrity of the pressure boundary. This condition is reported pursuant 10 CFR 50.73(a)(2)(ii).

B. Initial Condition

The reactor was in OPERATIONAL CONDITION 5 with reactor temperature at approximately 80 degrees F at the time of discovery.

C. Description of Occurrence

On October 27, 1993, plant personnel identified a leak from the TW associated with temperature element B33N028B. The component is located on the suction side of reactor recirculation water pump "B". The instrument located in this TW is associated with the reactor recirculation water pump thermal interlocks. Chemical analysis, along with a pressure drop test, confirmed the TW had a through-wall leak.

The component was replaced and retested satisfactorily. The remaining TWs and associated temperature element installations were visually inspected and no evidence of leakage was identified.

Non-destructive examinations of the failed TW were performed. A circumferential crack around approximately 3/4 of the TW was identified.

D. Apparent Cause

An analysis of the failed component was to be performed; however the component was lost following Refueling Outage 6.

Based on reviews of the design of the TW, visual inspection of the failed component and non destructive examination, two possible failure mechanisms were identified, flow induced high cycle low stress fatigue and IGSCC.

Based on the configuration of the crack, the design and materials of the TW (316L Stainless Steel) and a review of similar flow-induced fatigue failures at GGNS, it was concluded that the most probable cause of failure is flow-induced high cycle low stress fatigue.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)	PAGE (3)
Grand Gulf Nuclear Station	05000-416	93-014-01	3 OF 03

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

E. Corrective Actions

The current design of TW installation will be reviewed to determine if redesign is warranted.

F. Safety Assessment

The TW is designed to maintain its pressure boundary integrity. A complete failure of the TW could have possibly resulted in a 0.148 sq. inch opening in the pressure boundary. Since a complete failure of the TW would result in the loss of reactor coolant this condition can be compared to analyses described in the SAR to assess the safety significance.

The containment and reactor coolant systems are designed to withstand the consequences of small breaks in the pressure boundary such as a failure of a TW. It is assumed that the area of such break is less than 0.1 sq. feet. The opening, as stated above, as a result of a complete failure of the TW is well within the bounds of the conditions that were analyzed in the SAR for containment response to a small primary system line break. Therefore, the failed TW did not place the plant in an unanalyzed condition regarding containment design. Plant personnel concluded the worst case leakage would occur with a complete circumferential crack. This amount of leakage would have been less than 150 gpm which is bounded by the GGNS accident analysis.

ECCS systems were available to provide makeup to vessel inventory. The failure did not degrade any of the systems needed for emergency core cooling. Further, the SAR analyzes the effects of small breaks in the reactor recirculation system considering appropriate design criteria such as single failures. These analyses show that peak fuel temperature remains well below regulatory limits. Therefore this failure did not compromise the health and safety of the public.

The growth rate of the crack is dependent on the cyclic stress applied to the TW. Evaluations indicated that the stress that resulted from the recirculation flow was approximately 19 percent of the endurance strength of the TW material. Based on the stress evaluation and review of past leakage rate data, it was concluded that the growth rate of the crack was low. The remaining TWs are not presently leaking and the crack in the failed TW appeared to have gradually worsened to the condition observed during the inspection. Any leakage occurring in the three remaining original thermowells would be detected by the reactor coolant leakage detection system. The action for unacceptable leakage requires a Technical Specification shutdown. Therefore, no operating restrictions are warranted.

G. Additional Information

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