



ENTERGY

Entergy Operations, Inc.

P.O. Box 756

Port Gibson, MS. 39150

Tel 601 437 2800

April 15, 1994

C. R. Hutchinson

Vice President

Operations

Grand Gulf Nuclear Station

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

Attention: Document Control Desk

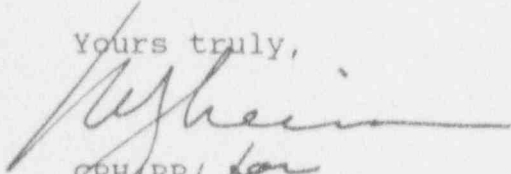
SUBJECT: Grand Gulf Nuclear Station  
Unit 1  
Docket No. 50-416  
License No. NPF-29  
Pressure Boundary Leakage Due to Thermowell Failure  
LER 93-014-01

GNRO-94/00050

Gentlemen:

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

Yours truly,

  
CRH/RR/  
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)  
Mr. H. L. Thomas (w/o)

Mr. Stewart D. Ebnetter (w/a)  
Regional Administrator  
U.S. Nuclear Regulatory Commission  
Region II  
101 Marietta St., N.W., Suite 2900  
Atlanta, Georgia 30323

Mr. P. W. O'Connor  
Office of Nuclear Reactor Regulation  
U.S. Nuclear Regulatory Commission  
Mail Stop 13H3  
Washington, D.C. 20555

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NRC FORM 366 (5-92)				U.S. NUCLEAR REGULATORY COMMISSION				APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95			
<b>LICENSEE EVENT REPORT (LER)</b>								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 (MNB 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) <b>Grand Gulf Nuclear Station</b>								DOCKET NUMBER (2) <b>05000-416</b>		PAGE (3) <b>01 of 03</b>	
TITLE (4) <b>Update to Pressure Boundary Leakage Due to Thermowell Failure</b>											
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	
11	02	93	93	014	01	04	15	94	N/A	05000	
									FACILITY NAME	DOCKET NUMBER	
									N/A	05000	
OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR 5 (Check one or more (11))									
5		20.402(b)				20.405(c)				50.73(a)(2)(iv) 73.71(b)	
POWER LEVEL (10)		20.405(a)(1)(i)				50.36(c)(1)				50.73(a)(2)(v) 73.71(c)	
000		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 <b>Riley Ruffin / Licensing Specialist</b>						TELEPHONE NUMBER (Include Area Code) <b>601-437-2167</b>					
COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)											
CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPROS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPROS	
X	AD	TW	R369	Y							
SUPPLEMENTAL REPORT EXPECTED (14)											
YES (If yes, complete EXPECTED SUBMISSION DATE)				X NO				EXPECTED SUBMISSION DATE (15)		MONTH DAY YEAR	
ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)											
<p>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.</p>											

NRC FORM 308A (5-82)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95	
<p align="center"><b>LICENSEE EVENT REPORT (LER)</b> <b>TEXT CONTINUATION</b></p>		<p>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.</p>	
		FACILITY NAME (1) <b>Grand Gulf Nuclear Station</b>	DOCKET NUMBER (2) <b>05000-416</b>

TEXT (If more space is required, use additional copies of NRC Form 308A) (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.

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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 (EIIIS) codes are identified in the text within brackets [ ].