Entergy

Entergy Operations, Inc.

William A. Eaton

December 2, 1999

U.S. Nuclear Regulatory Commission
Mail Stop P1-37
Washington, D.C. 20555

Attention: Document Control Desk

SUBJECT: Grand Gulf Nuclear Station
Docket No. 50-416
License No. NPF-29
Main Steam Lines Exceeded Leakage Limits
LER 1999-006-00

GNRO-99/00091

Gentlemen:

Attached is Licensee Event Report (LER) 1999-006-00 which is a interim report. Should you have any questions or require additional information regarding the contents of this report, please contact the licensing representative listed on the attached LER.

Yours truly,

WAE/CEB
attachment

cc: Ms. J. L. Dixon-Herrity, GGNS Senior Resident (w/a)
Mr. D. E. Levanway (Wise Carter)
Mr. L. J. Smith (Wise Carter) (w/a)
Mr. N. S. Reynolds (w/a)
Mr. H. L. Thomas (w/o)
Mr. J. W. Yelverton (w/a)

Mr. Ellis W. Merschoff (w/a)
Regional Administrator
U.S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive Suite 400
Arlington, TX 76011

Mr. S. P. Sekerak, NRR/DLPM/PD IV-I (W/2)
U.S. Nuclear Regulatory Commission
One White Flint North, Mail Stop 04-D3
11555 Rockville Pike
Rockville, MD 20852-2378

IE22

PDR ADOCK 05000416

bcc:

Ms. C. W. Gunn (w/3)
Required Reading Coordinator (w/a)
SRC Secretary (w/a)
TRG Secretary (w/a)
PSRC (w/a)
File (LCTS) (w/a)
File (RPTS) (w/a)
File (NS&RA) (w/a)
File (Central) (w/a) (//)

INPO Records Center (w/a)
700 Galleria Parkway
Atlanta, Georgia 30339-5957

Mr. J. Sipek (w/a)
Illinois Power Company
Clinton Power Station Mail Stop V-920
P.O. Box 678
Clinton, Illinois 61727

ccmail:

* Mr. W. B. Abraham
Mr. G. R. Ashley (ANO - LIC)
Mr. J. G. Booth
* Mr. C. A. Bottemiller
Mr. P. W. Brewer
Mr. C. E. Brooks
Mr. O. P. Bulich (RB - OE)
Mr. S. A. Burris
Mr. W. C. Cade
Mr. R. W. Carroll
Mr. K. E. Christian
Mr. W. F. Clark
Mr. D. G. Cupstid
Mr. J. P. Czaika
* Mr. L. F. Daughtery
Mr. G. G. Davie (W3 - LIC)
Ms. L. F. Davis
Mr. D. R. Denton (ANO)
* Mr. C. M. Dugger (W3 - VP)
* Mr. W. A. Eaton
Mr. R. D. Edington (RBS – VP)
Mr. C. W. Ellsaesser
Mr. J. L. Ensley (ESI)
* Mr. E. C. Ewing (W3 – LIC)
Mr. D. R. Franklin
Mr. T. J. Gaudet (W3 – LIC)
Ms. C. W. Gunn
Mr. E. D. Harris
Mr. C. D. Holifield
Mr. M. G. Hurley
* Mr. C. R. Hutchinson (ANO - VP)
IEA (ANO - OE)
Mr. R. D. Jacks
* Mr. D. E. James (ANO - LIC)
* Mr. R. S. Johnson
* Mr. M. L. Jones
Mr. R. J. King (RB - LIC)
Mr. M. A. Krupa (ECH – LIC)
* Mr. C. W. Lambert
* Mr. M. J. Larson
* Ms. J. M. Manzella (W3 – QP)
* Mr. J. R. McGaha (ECH)
* Mr. T. O. McIntyre
Mr. B. Mitchell
Mr. R. V. Moomaw
Ms. L. A. Patterson
Mr. C. H. Quick
* Mr. S. D. Reeves
Mr.C. M. Renfroe
* Mr. J. C. Roberts
Mr. J. L. Robertson
Mr. W. M. Shelly
Mr. C. D. Stafford
Mr. G. D. Swords
* Mr. R. L. Thomas (ECH)
Mr. T. H. Thurmon
Mr. J. D. Vandergrift (ANO – LIC)
Mr. J. E. Venable
* Mr. A. D. Wells (RB)
Mr. R. A. Wilson

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001
LICENSEE EVENT REPORT (LER)	Estimated burden per response to comply with this mandatory information collection request 50.0 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), 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. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Grand Gulf Nuclear Station	05000-416	1 of 5

TITLE (4)
Main Steam Lines Exceeded Leakage Limits

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	1999	1999	-- 006 --	00	12	02	1999	N/A	05000

OPERATING MODE (9)	5	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	00		20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		Specify in Abstract below or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Charles E. Brooks / Senior Licensing Specialist	601-437-6555

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	SJ	ISV	1038	Y	X				

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X	YES (If yes, complete EXPECTED SUBMISSION DATE)		NO		02	28	2000

ABSTRACT (Limit to 1400 spaces, i. e., approximately 15 single-spaced typewritten lines) (16)

On November 2 and November 13, 1999, with the reactor shutdown, a review of data from Main Steam Line (MSL) Local Leak Rate Testing (LLRT) determined that the as-found leakage through both the inboard and outboard Main Steam Isolation Valves (MSIVs) in the A and C MSLs exceeded the acceptance limit. TS 3.6.1.3 specifies a total leakage limit of 100 standard cubic feet per hour (scfh), equivalent to 47,200 standard cubic centimeters per minute (sccm) for all four MSLs. The acceptance leakage limit for individual MSIVs is 11,800 sccm (25 scfh). The as-found leakage rate through the B21F022A (inboard MSIV on the A MSL) was approximately 208,602 sccm (440.7 scfh) and unquantifiable through both C MSL MSIVs due to excessive leakage beyond the measuring capability of the test equipment. Total leakage through both the A and C MSLs exceeded the total allowable leakage limit specified in TS 3.6.1.3 for all four MSLs.

Since the MSIVs are designed to automatically close following a level 1 reactor water level signal indicative of a loss of coolant accident (LOCA), total leakage would be expected to have been attenuated to the leakage rate of the valve in each affected MSL with the minimum leakage (minimum pathway leakage). Additionally, the Main Steam Isolation Valve Leakage Control System in association with the Main Steam Shutoff Valves (MSSVs) were available to limit radiological releases to the environment. The MSSVs successfully passed leak rate testing. All MSIVs have been appropriately reworked and retested satisfactorily.

Although the design basis MSIV leakage limits would have been exceeded given a postulated LOCA, there were no actual safety consequences or compromises to public health and safety as a result of this event.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Grand Gulf Nuclear Station	05000-416	1999	006	00	2 OF 5

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

A. Reportable Occurrence

During Local Leak Rate Testing, the leakage rates of six main steam isolation valves resulted in exceeding the allowable leakage permitted by the Technical Specifications. Of the six MSIVs [EIIS Codes: JM, SIV] with as-found leakage exceeding the allowable limits, two were in the A MSL, two were in the C MSL, and one was in each of the remaining steam lines.

Telephone notifications were made to the NRC's Emergency Notification System on November 2 and November 13, 1999, reporting this condition pursuant to 10CFR50.72(b)(2)(i) - as a condition found while the reactor was shutdown, that if it had been found while the reactor was in operation, would have resulted in the nuclear power plant, including its principle safety barriers, being seriously degraded or in an unanalyzed condition, that significantly compromises plant safety.

This event also meets the following 30-day follow-up reporting criteria:

10CFR50.73(a)(2)(ii) in that MSL containment penetration leakage through both the inboard and outboard MSIVs was in excess of the total "as-found" maximum pathway leakage allowed by theTechnical Specification and represents a condition that was outside the design basis of the plant.

B. Initial Conditions

The plant was in Operational Condition 5, Refueling, during performance of the tests.

C. Description of Occurrence

Technical Specification 3.6.1.3 requires that the leakage rate through all four main steam lines be limited to a combined leakage rate of less than or equal to 100 scfh (47,200 sccm). Following performance of Local Leak Rate Testing of all eight MSIVs, the leakage rates for six of the valves exceeded both the Technical Specification leakage limit for total MSL leakage and the acceptance limit of 11,800 sccm (25 scfh) for each individual MSIV. Of this population of six MSIVs, the leakage through four valves could not be quantified due to excessive leakage beyond the capability of the test equipment to pressurize the test volume to the required test pressure. Thus, the leakage is considered to have exceeded total MSL leakage of 100 scfh (47,200 sccm). As-found data for the individual MSIVs is provided in Section G of this report. Section G, Table 1 provides an information only schematic of the valve configurations discussed.

As a result of this event, Condition Report GGCR 1999-1653, which references other related condition reports, and a Root Cause Analysis were initiated.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Grand Gulf Nuclear Station	05000-416	1999	006	00	3 OF 5

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

D. Apparent Cause

The excess "as-found" MSL leakage was the result of various mechanical abnormalities internal to the valves. In general, these abnormalities are attributed to normal wear, loose internal parts, and oxide/corrosion build up on the seating surfaces, resulting in failure of the valve to properly seat. The leakage rate for the "A" inboard MSIV was in excess of 100 scfh (47,200 sccm), which is the Technical Specification allowable leakage limit for all four main steam lines. Both MSIVs on the C MSL had leakage rates that were unquantifiable. For the B and D MSLs, although found to have one failed MSIV, the redundant MSIV in the B and D steam lines passed their leak rate testing and would have mitigated leakage through the respective MSLs. Additional details describing the various mechanical abnormalities will be provided in a supplement to this LER. The supplemental report is expected to be submitted by February 28, 2000.

E. Corrective Actions

1. The six MSIVs that initially failed their LLRT have been refurbished and successfully passed subsequent leak rate testing. Additionally, the two remaining MSIVs that initially passed their LLRT have also been conservatively refurbished and re-certified. This corrective action has been completed.

2. An evaluation is being performed to determine additional corrective actions to prevent recurrence of this condition. Any actions determined to be appropriate will be taken prior to startup from the next refueling outage (RFO-11).

3. Supplement the information contained in this LER upon completion of the root cause analysis.

F. Safety Assessment

The LOCA dose calculation is the only dose calculation that explicitly considers the MSIV leak rate; other events, such as the main steam line break outside containment, credit the MSIV isolation only since the calculated doses are not particularly sensitive to small leakage rates compared to the initial large release. The LOCA dose analysis assumes a single active failure in association with the recirculation line break which is currently assumed to be the failure of an MSIV to close. A unquantifiable leak rate on the B21F028A, B21F022C, B21F028C and the failure of the inboard MSL A MSIV, B21F022A, would represent a direct leakage path from the reactor vessel (where the source term release is occurring) to the Turbine Building, which is then released unfiltered to the environment since the Turbine Building is not part of Secondary Containment. Explicit calculations have not been performed, but it is expected that this bypass release path would result in offsite doses greater than the limits in 10CFR100 and control room doses greater than the limits in 10CFR50, Appendix A, Criterion 19.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Grand Gulf Nuclear Station	05000-416	1999	006	00	4 OF 5

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

Safety Assessment (Continued)

For severe accidents involving large radiological releases, the actual consequences would have been substantially mitigated by the MSIV Leakage Control System (MSIV-LCS). This safety related system supplements the isolation function of the MSIVs by processing the fission products (by directing it into the secondary containment for processing by the Standby Gas Treatment System) that could leak through the closed MSIVs after a Design Basis Accident (DBA) loss of coolant accident (LOCA). The MSIV LCS consists of two independent subsystems: an inboard subsystem, which is connected between the inboard and outboard MSIVs; and an outboard subsystem, which is connected immediately downstream of the outboard MSIVs. Each subsystem is capable of processing leakage from MSIVs following a DBA LOCA. The outboard subsystem is further isolated from the environment by the Main Steam Shutoff Valves (MSSVs) B21F098A, B, C, and D. These valves are safety related with emergency backup power provided by the Division II ESF bus and are remotely operated from the control room by procedure (System Operating Instruction 04-1-01-E32-1) to support operation of the outboard MSIV-LCS. During the current refueling outage, these valves successfully passed Local Leak Rate Testing such that the outboard MSIV-LCS function was capable of substantially limiting the radiological consequences of MSIV leakage

G. Additional Information

Main Steam Line Isolation Valve As-Found Leak Rate Testing Results
Units are Standard Cubic Centimeters Per Minute (sccm)

Valve Description		Test Result	As-Found Leakage (sccm)
B21F022A	MSL A Inboard	Failed	208,602
B21F022B	MSL B Inboard	Failed	15,300
B21F022C	MSL C Inboard	Failed	Unquantifiable
B21F022D	MSL D Inboard	Passed	11,300
B21F028A	MSL A Outboard	Failed	Unquantifiable
B21F028B	MSL B Outboard	Passed	200
B21F028C	MSL C Outboard	Failed	Unquantifiable
B21F028D	MSL D Outboard	Failed	Unquantifiable

As discussed in the safety assessment (Section F), the Main Steam Shutoff Valves were also leak rate tested. The as-left leakage rate data is as follows:

B21F098A – 1497 sccm
B21F098B – 251 sccm
B21F098C – 2254 sccm
B21F098D – 2002 sccm

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

NRC FORM 366A
(6-1998)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Grand Gulf Nuclear Station	05000-416	1999	006	00	5 OF 5

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

G. (Continued)

Figure 1