ENTERGY

Entergy Operations, Inc.
P.O. Box 756
Port Gibson, MS 39150
Tel 601 437 2800

C. R. Hutchinson
Vice President
Operations
Grand Gulf Nuclear Station

June 14, 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
Failure to Adequately Incorporate Existing Design Into GGNS Operating and Emergency Procedures
LER 94-005-00

GNRO-94/00087

Gentlemen:

Attached is Licensee Event Report (LER) 94-005 which is a final report.

Yours truly,

C. R. Hutchinson

CRH/CDH/
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. Ebneter (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

200069

IE22
1/1

Attachment to GNRO-94/00087

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)
Failure to Adequately Incorporate Existing Design Into GGNS Operating and Emergency Procedures

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
05	17	94	94	005	00	06	14	94	N/A	05000

OPERATING MODE (9)	5	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)	000		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)
Charles Holifield / Licensing Engineer	601-437-6439

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)

During containment pressurization for the Integrated Leak Rate Test during Refueling Outage Six, substantial leakage was found and was determined to be from the Drywell Fission Product Monitor and Post Accident Sample System sample lines. The lines were isolated and the Integrated Leak Rate Test was completed on November 21, 1993. A Material Non-Conformance Report was written to document the leakage problem. During the evaluation, it was discovered that there was no clear guidance to close the containment isolation valves for the Drywell Fission Product Monitor and Post Accident Sample System when required post-accident. This condition is reportable per 10 CFR 50.73(a)(2)(ii)(C).

The apparent cause of the condition is failure to adequately incorporate the existing design into GGNS operating and emergency procedures. The dual function of the line supplying the Drywell Fission Product Monitor and Post Accident Sample Systems apparently caused a procedural oversight.

Corrective actions include generating a Minor Change Package to incorporate the auto-isolate feature on the subject penetration isolation valves and evaluating other potential containment post-accident bypass leak paths.

Conservative analyses based on as-found leakage rates confirmed that the safety and health of the general public were not compromised by this condition.

NRC FORM 366A (5-92) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95		
LICENSEE EVENT REPORT (LER) TEXT CONTINUATION	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)	LER NUMBER (6)	PAGE (3)
Grand Gulf Nuclear Station, Unit 1	05000-416	94-005-00	2 OF 04

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

A. Reportable Occurrence

During performance of 10 CFR 50, Appendix J, Primary Containment [NG] Integrated Leak Rate Test (ILRT), leakage was discovered in the Drywell Fission Product Monitor Sample (DFPM) [IJ] panel and the Post Accident Sample System (PASS) [IP]. The associated containment isolation valves were subsequently closed. This event resulted in discovery that closure of the isolation valves was not adequately covered in GGNS operating and emergency procedures. After extensive evaluation, this condition was deemed reportable per 10 CFR 50.73(a)(2)(ii)(C) on May 17, 1994.

B. Initial Conditions

At the time of occurrence, the reactor was in OPERATIONAL CONDITION 5 with reactor water temperature at approximately 95 degrees F and depressurized to primary containment atmospheric pressure. The unit was in its sixth refueling outage (RFO6). Primary containment was pressurized in preparation for performance of an ILRT.

C. Description of Occurrence

On September 23, 1993, GGNS modified the Local Leak Rate Test (LLRT) program by deleting the requirements to perform Appendix J, Type C LLRT on forty-three containment isolation valves postulated to remain open post-accident. These valves would be left open during subsequent ILRT surveillances. The changes were performed in accordance with an approved Safety Evaluation (SE). Two of the penetrations referenced in the SE provide the recirculation path for the DFPM and the post accident sampling of the drywell atmosphere by the PASS system. The supply line is isolated by remote-manual-operated containment isolation valves D23F591-B (inboard) and D23F592-A (outboard) in penetration 109A. The return line is isolated by remote-manual-operated containment isolation valves D23F593-B (inboard) and D23F594-A (outboard) in penetration 109B.

The SE stated that the valves need not be LLRTed based on several 10 CFR 50 Appendix J qualifiers which include:

- valves do not auto-isolate on a Loss of Coolant Accident (LOCA); remain open in all modes of operation
- lines form a closed loop outside containment and will be tested periodically
- lines are small ($^3/_4$")

NRC FORM 366A (5-92) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95
LICENSEE EVENT REPORT (LER) TEXT CONTINUATION	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)	LER NUMBER (6)	PAGE (3)
Grand Gulf Nuclear Station, Unit 1	05000-416	94-005-00	3 OF 04

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

As a result of the change, the ILRT line-up was modified to allow the containment isolation valves to remain open during the ILRT performed during RFO6.

During containment pressurization in preparation for the ILRT on November 20, 1993, substantial leakage was found and determined to be from the DFPM and PASS sample lines. The DFPM and PASS were isolated from the containment by closing the containment isolation valves and the ILRT was successfully completed on November 21, 1993. A Material Non-Conformance Report (MNCR) was written to document the leakage problem. Subsequent evaluation per the MNCR resulted in discovery that there was no clear guidance to close the valves when required post-accident.

D. Apparent Cause

The apparent cause of the condition is failure to adequately incorporate the existing design into GGNS operating and emergency procedures. The dual function of the line supplying the DFPM and PASS systems apparently caused a procedural oversight. The most appropriate emergency response action for these isolation valves is that they be closed shortly after an accident and re-opened when a PASS sample is required. No clear procedural guidance was provided to accomplish this action.

Due to the length of time since the dual function design was implemented (prior to plant licensing), the root cause for unclear procedural guidance is indeterminate. This design was included in licensing basis documents to accomplish the PASS function, however adequate procedural guidance was not developed to support the PASS design. In hindsight, other design features, such as auto-isolation, could have been incorporated to better accomplish this action.

E. Corrective Actions

1) Operations issued a standing order directing operators to manually close these valves in the event of a LOCA.

2) Nuclear Plant Engineering is generating a Minor Change Package (MCP) to incorporate the auto-isolate feature on penetration 109A and 109B isolation valves.

3) As an interim measure pending final implementation of the MCP, the DFPM fittings were repaired to minimize leakage and the PASS drywell atmosphere sample line was cut and capped.

NRC FORM 366A (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)	PAGE (3)
Grand Gulf Nuclear Station, Unit 1	05000-416	94-005-00	4 OF 04

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

4) Plant Staff restored the requirement to perform LLRTs on the 109A and 109B isolation valves and performed successful LLRTs on them on November 21, 1993.

5) In order to confirm that other similar situations do not exist, NPE will evaluate other potential containment post-accident bypass leak paths.

F. Safety Assessment

There was no decrease in the ability of primary containment, in conjunction with available accident mitigation systems, to prevent offsite and control room doses from exceeding 10 CFR 100 and GDC 19 limits.

A conservative analysis based on initial ILRT leakage data indicated that the as-found ILRT leakage rate, when applied to the LOCA dose model, would result in doses lower than those from the current DBA dose analyses, except in one instance. The 2 hour thyroid dose at the Site Boundary would have been 34.78 REM for the as-found condition as compared to 18.84 REM in the DBA analyses. However, this is much lower than the acceptable limits of 10 CFR 100.

These calculations assume the containment isolation valves are shut at 20 minutes into the event. None of the cases analyzed approached 10 CFR 100 or GDC 19 limits. Manual closure at 20 minutes into a DBA event is consistent with regulatory guidance and similar GGNS LOCA analysis assumptions.

As discussed above, the safety and health of the general public were not compromised by this event.

G. Additional Information

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