UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 14, 2001

Mr. William A. Eaton
Vice President, Operations GGNS
Entergy Operations, Inc.
P. O. Box 756
Port Gibson, MS 39150

SUBJECT: GRAND GULF NUCLEAR STATION (GGNS), UNIT 1 - ISSUANCE OF AMENDMENT RE: FULL-SCOPE IMPLEMENTATION OF AN ALTERNATIVE ACCIDENT SOURCE TERM (TAC NO. MA8065)

Dear Mr. Eaton:

The Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 145 to Facility Operating License No. NPF-29 for the GGNS, Unit 1. This amendment consists of changes to the Faciltiy Operating License and Technical Specifications (TSs) in response to your application dated January 21, 2000, supplemented by letters dated June 29, September 1, October 26, and December 22, 2000, and February 22, 2001. The October 26, 2000, and the February 22, 2001, supplemental letters contained changes to the TS pages and to the operating license page submitted with the original application. The June 29, September 1, and December 22, 2000, supplemental letters provided clarifying information regarding the original application. These changes and clarifying information do not affect NRC's finding of no significant hazards consideration determination published in the *Federal Register* on March 22, 2000 (65 FR 15380).

The amendment provides for a full-scope implementation of the alternative source term (AST), as described in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors," and 10 CFR 50.67, "Accident source term."

In response to your request, the NRC staff completed its review of proposed changes to the GGNS TSs implementing the AST. On the basis of the staff's review of the GGNS analysis of the radiological consequences for three design-basis accidents (DBAs): the loss-of-coolant accident, the fuel-handling accident, and the control rod drop accident, and the staff's confirmatory assessment of the radiological consequences of these postulated DBA's, we find that the proposed changes to the GGNS TSs are acceptable. The staff's review is summarized in the Safety Evaluation also enclosed with this letter.

The staff recognizes the participation of GGNS as a rebaselining assessment facility, as a pilot plant, and as a member of the Nuclear Energy Institute Task Force supporting the development of the AST rule and related regulatory guidance associated with this important technical and regulatory issue.

The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice. The changes shall be implemented within 60 days from the date of issuance. You are requested to inform the staff by letter when these changes have been implemented at GGNS. If you have any questions regarding this subject, please contact me at 301-415-2623.

Sincerely,

S. Patrick Sekerak

S. Patrick Sekerak, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-416

Enclosures: 1. Amendment No. 145 to NPF-29
2. Safety Evaluation

cc w/encls: See next page

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY OPERATIONS, INC.

SYSTEM ENERGY RESOURCES, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

ENTERGY MISSISSIPPI, INC.

DOCKET NO. 50-416

GRAND GULF NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 145
License No. NPF-29

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated January 21, 2000, as supplemented by letters dated June 29, September 1, October 26, and December 22, 2000, and February 22, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-29 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 145 , are hereby incorporated into this license. Entergy Operations, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. Furthermore, the following paragraph of Facility Operating License No. NPF-29 is hereby amended to read as follows:

   2.C (38) Control Room Leak Rate (Section 6.2.6, SSER #6)

   EOI shall operate Grand Gulf Unit 1 during Modes 1 through 3 with an allowable control room leak rate not to exceed 2000 cfm (not including ingress / egress leakage of 10 cfm).

4. This license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A Gramm

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: March 14, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 145

FACILITY OPERATING LICENSE NO. NPF-29

DOCKET NO. 50-416

Replace the following pages of the Appendix A Technical Specifications and License with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
Technical Specifications	
1.0-3	1.0-3
1.0-3a	1.0-3a
3.3-73	3.3-73
3.3-74	3.3-74
3.3-75	3.3-75
3.3-76	3.3-76
3.6-17	3.6-17
3.6-44	3.6-44
3.7-6	3.7-6
3.7-7	3.7-7
3.7-8	3.7-8
3.7-9	3.7-9
3.7-10	3.7-10
3.7-11	3.7-11
3.8-18	3.8-18
3.8-19	3.8-19
3.8-20	3.8-20
3.8-31	3.8-31
3.8-32	3.8-32
3.8-33	3.8-33
3.8-40	3.8-40
5.0-12	5.0-12
5.0-13	5.0-13
License	
15	15

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	be those listed in Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989.
EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME	The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial movement of the associated turbine stop valve or the turbine control valve to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured, except for the breaker arc suppression time, which is not measured but is validated to conform to the manufacturer's design value.
FRACTION OF CORE BOILING BOUNDARY (FCBB)	The FCBB shall be the ratio of the power generated in the lower 4 feet of the active reactor core to the power required to produce bulk saturated boiling of the coolant entering the fuel channels. The core boiling boundary is the axial elevation of core average bulk saturation above the bottom of the active reactor core.
ISOLATION SYSTEM RESPONSE TIME	The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. The response time

(continued)

1.1 Definitions

ISOLATION SYSTEM RESPONSE TIME (continued)	may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
L_a	The maximum allowable primary containment leakage rate, L_a, shall be 0.682% of primary containment air weight per day at the calculated peak containment pressure (P_a).

(continued)

3.3 INSTRUMENTATION

3.3.7.1 Control Room Fresh Air (CRFA) System Instrumentation

LCO 3.3.7.1 The CRFA System instrumentation for manual isolation shall be OPERABLE.

APPLICABILITY: Modes 1, 2, and 3
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Place channel in trip.	24 hours
B. Required Action and associated Completion Time.	B.1 Close associated isolation dampers.	1 hour

Text Deleted

Table 3.3.7.1-1

Deleted

SURVEILLANCE REQUIREMENTS

------------------------------------NOTE------------------------------------

When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided CR isolation capability is maintained.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1.1	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify the isolation time of each MSIV is $\geq$ 3 seconds and $\leq$ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	18 months
SR 3.6.1.3.8	------------------NOTE------------------ Only required to be met in MODES 1, 2, and 3. -- Verify leakage rate through each main steam line is $\leq$ 100 scfh when tested at $\geq P_a$, and the total leakage rate through all four main steam lines is $\leq$ 250 scfh when tested at $\geq P_a$.	In accordance with 10 CFR 50, Appendix J, Testing Program
SR 3.6.1.3.9	------------------NOTE------------------ Only required to be met in MODES 1, 2, and 3. -- Verify combined leakage rate of 1 gpm times the total number of PCIVs through hydrostatically tested lines that penetrate the primary containment is not exceeded when these isolation valves are tested at $\geq 1.1\ P_a$.	In accordance with 10 CFR 50, Appendix J, Testing Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.3	Verify each standby gas treatment (SGT) subsystem will draw down the secondary containment to ≥ 0.25 inch of vacuum water gauge in ≤ 180 seconds.	18 months on a STAGGERED TEST BASIS
SR 3.6.4.1.4	Verify each SGT subsystem can maintain ≥ 0.266 inch of vacuum water gauge in the secondary containment for 1 hour at a flow rate ≤ 4000 cfm.	18 months on a STAGGERED TEST BASIS

3.7 PLANT SYSTEM

3.7.3 Control Room Fresh Air (CRFA) System

LCO 3.7.3 Two CRFA subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CRFA subsystem inoperable.	A.1 Restore CRFA subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during OPDRVs.	C.1 Place OPERABLE CRFA subsystem in isolation mode. OR C.2 Initiate action to suspend OPDRVs.	Immediately Immediately
D. Two CRFA subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two CRFA subsystems inoperable during OPDRVs.	E.1 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Operate each CRFA subsystem for $\geq$ 10 continuous hours with the heaters operating.	31 days
SR 3.7.3.2	Perform required CRFA filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.3.3	Verify each CRFA subsystem actuates on an actual or simulated initiation signal.	18 months

3.7 PLANT SYSTEMS

3.7.4 Control Room Air Conditioning (AC) System

LCO 3.7.4 Two control room AC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One control room AC subsystem inoperable.	A.1 Restore control room AC subsystem to OPERABLE status.	30 days
B. Two control room AC subsystems inoperable.	B.1 Verify control room area temperature ≤ 90°F. AND B.2 Restore one control room AC subsystem to OPERABLE status.	Once per 4 hours 7 days
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A not met during OPDRVs.	D.1	Place OPERABLE control room AC subsystem in operation.	Immediately
	OR		
	D.2	Initiate action to suspend OPDRVs.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition B not met during OPDRVs.	E.1 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify each control room AC subsystem has the capability to remove the assumed heat load.	18 months

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources — Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems — Shutdown"; and

b. One diesel generator (DG) capable of supplying one division of the Division 1 or 2 onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.8; and

c. One qualified circuit, other than the circuit in LCO 3.8.2.a, between the offsite transmission network and the Division 3 onsite Class 1E electrical power distribution subsystem, or the Division 3 DG capable of supplying the Division 3 onsite Class 1E AC electrical power distribution subsystem, when the Division 3 onsite Class 1E electrical power distribution subsystem is required by LCO 3.8.8.

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the primary or secondary containment.

ACTIONS

-----------------------------------NOTE-----------------------------------

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO Item a not met.	------------NOTE------------ Enter applicable Condition and Required Actions of LCO 3.8.8, when any required division is de-energized as a result of Condition A. ----------------------------	
	A.1 Declare affected required feature(s) with no offsite power available from a required circuit inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of recently irradiated fuel assemblies in the primary and secondary containment.	Immediately
	AND	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs).	Immediately
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2.4	Initiate action to restore required offsite power circuit to OPERABLE status.	Immediately
B. LCO Item b not met.	B.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	B.2	Suspend movement of recently irradiated fuel assemblies in primary and secondary containment.	Immediately
	AND		
	B.3	Initiate action to suspend OPDRVs.	Immediately
	AND		
	B.4	Initiate action to restore required DG to OPERABLE status.	Immediately
C. LCO Item c not met.	C.1	Declare High Pressure Core Spray System inoperable.	72 hours

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources — Shutdown

LCO 3.8.5 The following shall be OPERABLE:

a. One Class 1E DC electrical power subsystem capable of supplying one division of the Division 1 or 2 onsite Class 1E DC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems - Shutdown";

b. One Class 1E battery or battery charger, other than the DC electrical power subsystem in LCO 3.8.5.a, capable of supplying the remaining Division 1 or 2 onsite Class 1E DC electrical power distribution subsystem(s) when required by LCO 3.8.8; and

c. The Division 3 DC electrical power subsystem capable of supplying the Division 3 onsite Class 1E DC electrical power distribution subsystem, when the Division 3 onsite Class 1E DC electrical power distribution subsystem is required by LCO 3.8.8.

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the primary or secondary containment.

ACTIONS

------------------------------------NOTE------------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required battery charger inoperable.	------------NOTE------------ Entry into MODE 4 or 5, or commencing movement of recently irradiated fuel is not allowed, except entry into MODE 4 or 5 can be made as part of a unit shutdown. ---------------------------- A.1 Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	 1 hour AND Once per 8 hours thereafter
B. Required Action and associated Completion Time of Condition A not met.	B.1 Declare associated battery inoperable.	Immediately
C. One or more required DC electrical power subsystems inoperable for reasons other than Condition A.	C.1 Declare affected required feature(s) inoperable. OR C.2.1 Suspend CORE ALTERATIONS. AND	Immediately Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2.2 Suspend movement of recently irradiated fuel assemblies in the primary and secondary containment.	Immediately
	<u>AND</u>	
	C.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	<u>AND</u>	
	C.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	--------------------NOTE------------------- The following SRs are not required to be performed: SR 3.8.4.4, SR 3.8.4.6, SR 3.8.4.7, and SR 3.8.4.8. -- For DC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.4 SR 3.8.4.7 SR 3.8.4.2 SR 3.8.4.5 SR 3.8.4.8. SR 3.8.4.3 SR 3.8.4.6	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Distribution Systems – Shutdown

LCO 3.8.8 The necessary portions of the Division 1, Division 2, and Division 3 AC and DC electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the primary or secondary containment.

ACTIONS

------------------------------------NOTE------------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC or DC electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of recently irradiated fuel assemblies in the primary and secondary containment.	Immediately
	<u>AND</u>	

(continued)

5.5 Programs and Manuals (continued)

5.5.7 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2.

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1975 at the system flowrate specified below ± 10%:

ESF Ventilation System	Flowrate
SGTS	4000 cfm
CRFA	4000 cfm

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1975 at the system flowrate specified below ± 10%:

ESF Ventilation System	Flowrate
SGTS	4000 cfm

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and the relative humidity specified below:

ESF Ventilation System	Penetration	RH
SGTS	0.5%	70%

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers (if used) is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1975 at the system flowrate specified below ± 10%:

ESF Ventilation System	Delta P	Flowrate
SGTS	9.2" WG	4000 cfm
CRFA	7.2" WG	4000 cfm

e. Demonstrate that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ANSI N510-1975 (except for the phase balance criteria stated in Section 14.2.3):

ESF Ventilation System	Wattage
SGTS	48 ± 5.0 kW
CRFA	20.7 ± 2.1 kW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the main condenser offgas treatment system and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

a. The limits for concentrations of hydrogen in the main condenser offgas treatment system and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

(continued)

(a) Include an emergency override of the test mode of the Division 3 HPCS diesel generator to permit response to emergency signals and to return the control of the diesel generator to the emergency standby mode. (Item No. 333, T.S. 4.8.1.1.2.d.12.b)

(b) Provide the second level undervoltage protection for Division 3 power supply (Item No. 373, T.S. Table 3.3.3-2).

(c) Incorporate a bypass or coincident logic in all Division 1 and 2 diesel generator protective trips, except for trips on diesel engine overspeed and generator differential current (Item No. 808, T.S. 4.8.1.1.2.d.16.d).

(38) Control Room Leak Rate (Section 6.2.6, SSER #6)

EOI shall operate Grand Gulf Unit 1 during Modes 1 through 3 with an allowable control room leak rate not to exceed 2000 cfm (not including ingress/egress leakage of 10 cfm).

(39) Temporary Secondary Containment Boundary Change

For a period of time not to exceed 144 cumulative hours, the provisions of Specification 3/4.6.6.1 may be applied to the railroad bay area including the exterior railroad bay door on the auxiliary building in lieu of the present secondary containment boundaries that isolate the railroad bay area. While the railroad bay area is being used as a secondary containment boundary, the railroad bay door may be opened for the purpose of moving trucks in and out provided the four hour limitation in ACTION a of Technical Specification 3.6.6.1 is reduced to one hour. A fire watch shall be established in the railroad bay area while the door is being used as a secondary containment boundary.

(40) Temporary Ultimate Heat Sink Change

With the plant in OPERATIONAL condition 4, SSW cooling tower basin A may be considered OPERABLE in accordance with Technical Specification 3.7.1.3 with less than a 30 day supply of water (without makeup) during the time that SSW basin B is drained to replace its associated service water pump provided:

(a) SSW basin A water level is maintained greater than or equal to 87".

(b) At least two sources of water (other than normal makeup with one source not dependent on offsite power) are available for makeup to SSW basin A.

This license condition may remain in effect until plant startup following the outage scheduled for fall 1985.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 145 TO FACILITY OPERATING LICENSE NO. NPF-29

ENTERGY OPERATIONS, INC., ET AL.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By letter dated January 21, 2000, as supplemented by letters dated June 29, September 1, October 26, and December 22, 2000, and February 22, 2001, Entergy Operations, Inc., et al. (the licensee) submitted a request for changes to the Grand Gulf Nuclear Station, Unit 1 (GGNS), Technical Specifications (TSs). The changes would revise the TSs to provide for a full-scope implementation of the alternative source term (AST), as described in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors," and 10 CFR 50.67, "Accident source term." For this license amendment request implementing the full-scope AST, the licensee reanalyzed and submitted the radiological consequences for three design-basis accidents (DBAs): the loss-of-coolant accident (LOCA), the fuel-handling accident (FHA), and the control rod drop accident (CRDA).

2.0 BACKGROUND

As the lead pilot plant application for implementing an AST at operating nuclear power plants, the licensee submitted this amendment with the endorsement of the Nuclear Energy Institute. In 1998, the staff used GGNS as a rebaselining plant for evaluating the impact of implementing the AST at operating nuclear plants as described in SECY 96-242, "Use of the NUREG-1465 Source Term at Operating Reactors." The results and findings from the rebaselining study for implementing the AST at operating reactors (the GGNS, Surry, and Zion stations) were provided in SECY 98-154, "Results of the Revised (NUREG-1465) Source Term Rebaselining for Operating Reactors."

Specifically, the licensee requested that:

1. TS 1.1, "Definitions," be amended to reference new dose conversion factors and to increase the maximum allowable primary containment leakage rate from 0.437 percent to 0.682 percent of primary containment air weight per day.

2. TS Bases 3.3.6.1, 3.3.6.2, 3.6.1.3, 3.6.4.1, 3.6.4.2, and 3.6.4.3 be revised to reflect the new interpretation of the term "recently irradiated fuel" as fuel which has been irradiated in the reactor within the previous 24 hours instead of within eight days.

3. TS 3.3.7.1, "Control Room Fresh Air System Instrumentation," be amended to delete all automatic control room isolation features from the scope of the GGNS TSs. This revision would delete all automatic isolation input signals from Table 3.3.7.1-1, delete Surveillance Requirements (SRs) 3.3.7.1.1 through 3.3.7.1.5, and revise Limiting Condition for Operation (LCO) 3.3.7.1 and its associated ACTION items. The only safety function required of the control room fresh air (CRFA) system would be manual control room isolation.

4. SR 3.6.1.3.8, "Main Steam Isolation Valve (MSIV) Leakage Rate," be amended to increase the maximum allowable leak rate to less than or equal to 100 standard cubic feet per hour (scfh) per main steam line (MSL) with a total leak rate through all four MSLs of less than or equal to 250 scfh (from less than or equal to 100 scfh through all four MSLs.)

5. SR 3.6.4.1.3, "Secondary Containment Drawdown," be amended to increase the maximum allowable drawdown time from 120 seconds to 180 seconds.

6. TS 3.7.3, "Control Room Fresh Air (CRFA) System," and its corresponding Bases Section B 3.7.3 be amended to address the sole isolation function of the CRFA system and the removal of radioactive aerosol by high-efficiency particulate air (HEPA) filters in the CRFA system. No credit for iodine removal by the CRFA system charcoal adsorbers would be taken for the purpose of the DBA radiological consequence analyses. This requested amendment would also delete the fuel movement and core alteration periods from the Applicability based on the revised fuel-handling accident analysis. The licensee stated that neither the CRFA system isolation nor filtration by the charcoal adsorbers or HEPA filters would be needed to meet the relevant dose criteria.

7. The Applicability Statement of TS 3.7.4, "Control Room Air Conditioning (AC) System," be amended to no longer require the Limiting Condition for Operation (LCO) to be met during fuel movement or CORE ALTERATIONS. The CONDITIONS, REQUIRED ACTIONS, and COMPLETION TIMES in the ACTIONS would also be modified accordingly.

8. TS 3.8.2, 3.8.5, and 3.8.8 be amended to modify the Applicability Statements of these LCOs to read "During movement of *recently* irradiated fuel assemblies..." instead of "During movement of irradiated fuel assemblies..." For the current fuel cycle, the licensee proposed this term to be defined as those fuel assemblies that have been in a critical reactor core within the previous 24 hours. The REQUIRED ACTIONS in the ACTIONS would be also modified accordingly.

In addition, the licensee requested to amend GGNS License Condition 2.C (38), "Control Room Leak Rate," to increase the maximum allowable control room leak rate during Modes 1 through 3 from 590 cubic feet per minute (cfm) to 2000 cfm, not including the 10 cfm inleakage attributable to control room ingress / egress consistent with Standard Review Plan Section 6.4 guidance.

3.0 EVALUATION

3.1 LOCA

The current radiological consequence analysis for the postulated LOCA using Technical Information Document (TID)-14844 source term is provided in the GGNS Updated Final Safety Analysis Report (UFSAR), Section 15.6. To demonstrate that the GGNS engineered safety features (ESFs) are designed to mitigate the radiological consequences of a LOCA and will remain adequate after implementation of this license amendment, the licensee reevaluated the offsite and control room radiological consequences of the postulated LOCA. The licensee has implemented the AST in this reevaluation. The licensee submitted the results of its offsite and control room dose calculations (see Table 1). In addition, the licensee provided a complete dose analysis and described the major assumptions and parameters used in its dose calculations, and the fission product transport, removal, and release models developed by the licensee. As documented in the submittals, the licensee has determined that after the implementation of the AST, the existing ESF systems at GGNS will still provide assurance that the total radiological consequences of the postulated LOCA at the exclusion area boundary (EAB), in the low population zone (LPZ), and in the control room meet the radiation dose criteria specified in 10 CFR 50.67.

The staff has reviewed the licensee's analysis and has performed an independent confirmatory radiological consequence dose calculation for the following three potential fission product release pathways after the postulated LOCA:

(1) MSIV leakage
(2) containment leakage
(3) post-LOCA leakage from ESF systems outside containment

The results of the staff's independent radiological consequence calculations are given in Table 1, alongside the licensee's results. The major parameters and assumptions used by the staff are listed in Tables 2 through 4.

3.1.1 MSIV Leakage Pathway

The GGNS MSIV leakage control system (LCS) is composed of independent inboard and outboard systems. The inboard system evacuates the leakage in the volume between the inboard and outboard MSIVs, while the outboard system draws the leakage from the volume downstream of the outboard MSIVs. The licensee assumed that the MSIV-LCS is manually actuated within 20 minutes of the postulated LOCA. This assumption is consistent with Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants." There are four MSLs at GGNS. Each MSL has an inboard MSIV and an outboard MSIV. These valves isolate the reactor coolant system in the event of a break in a steam line outside the primary containment, a design basis LOCA, or another event requiring containment isolation. Although the MSIVs are designed to provide a leak-tight barrier, it is recognized that some leakage may occur through these valves. The current GGNS TS limit for MSIV leakage is less than or equal to 100 scfh through all four MSLs. The licensee requested to increase the maximum allowable leak rate to less than or equal to 100 scfh per MSL with a total leak rate through all four MSLs of less than or equal to 250 scfh.

The staff assumed in its analysis that one of the two MSIVs in one MSL fails to close and all allowed leakage (100 scfh) is released directly to the environment through the remaining MSIV in this MSL for the first 20 minutes after the accident. No credit is provided for fission product deposition or holdup for decay in this steam line. Leakage on the remaining three MSLs (150 scfh) is directed into a volume between the closed inboard and outboard valves for the first 20 minutes. After 20 minutes, all MSIV leakage (250 scfh) is collected by the MSIV-LCS and discharged to the secondary containment (Enclosure Building) before release to the environment through the standby gas treatment system (SGTS).

3.1.1.1 Fission Product Transport in Drywell

The licensee assumed, and the staff agrees, that a large-break LOCA (LBLOCA) as a result of a double-ended guillotine pipe rupture in a recirculation suction line would be the most limiting LOCA with respect to the offsite and control room radiological consequences. The break releases reactor coolant to the drywell. No water injection from the emergency core cooling system (ECCS) is assumed, and the reactor water level drops below the core, exposing the reactor fuel. In GGNS License Amendment No. 143, issued on March 22, 2000, "Implementation of Alternative Source Term Limited Scope Application for the Timing of the Onset of Gap Activity Release," the staff evaluated the earliest time of fission products release (fuel gap activity release) from perforated fuel rods following a postulated LOCA and concluded that the minimum time would be no earlier than 120 seconds. The staff also assumed that all fission products are released directly to the drywell and leaked into the primary containment and into the MSLs, bypassing the suppression pool. The staff concludes that these assumptions are appropriate for the LBLOCA. For small-break LOCAs with operator actuation of an automatic depressurization system (ADS), most of the fission products would be released into the drywell through the pipe break and into the suppression pool through the ADS, where the fission products are removed.

As characterized in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," the gap and early in-vessel fission product releases terminate two hours after the postulated LOCA initiation. The staff assumed (as it did for Perry Nuclear Power Plant, Unit 1 (Perry), License Amendment No.103, issued on March 26, 1999, "Main Steam Line Leakage Requirements and Elimination of the Main Steam Isolation Valve Leakage Control System Implementing the Alternative Source Term") that the fission products are homogeneously distributed between the drywell and the primary containment two hours after accident initiation (both GGNS and Perry containment structures are Mark III type designs). This would require reflooding of the reactor vessel. Instead of trying to justify an all encompassing steaming rate due to this reflooding, the staff concludes that a substantial amount of fission products may end up in the primary containment as well as the drywell, and that mitigative features such as the SGTS need to be designed to accommodate a significant portion of the source term. For most of the risk significant cases, such as station blackout and transients, all the fission products are released directly to the primary containment via the safety relief valves. Waiting two hours after accident initiation to homogeneously mix the source term is acceptable for achieving an appropriate balance because the worst two hours are considered, instead of the first two hours used with the TID-14844 source term.

Confirmatory calculations performed by the staff showed that the radiological consequences are dependent upon the drywell bypass leakage prior to the termination of fission product release at two hours. Because of this sensitivity, the staff concludes, as it did for Perry, that

without relocation of reactor fuel to the lower head, the steaming rate of an intact core on the order of 3,000 cfm should be assumed for drywell bypass leakage. The staff's steaming rate prior to two hours is conservative in that it does not credit steaming due to relocation, cooling from alternative water sources (AWS), or the release of hydrogen gas.

The 3,000 cfm drywell bypass leakage rate is based upon LBLOCA analyses performed with the computer code MELCOR on a GGNS-type model. The results of these analyses showed: (a) no relocation of fuel below the reactor core plate, (b) the reactor water level decreased to a level below the core plate, and (c) an average steaming rate of approximately 2,800 cfm prior to quenching of the core at approximately 0.5 hours. Also, AWSs, such as the standby liquid control system, would not be available during station blackout sequences which comprise 96 percent of the core damage frequency for GGNS. Therefore, the staff concludes the use of 3,000 cfm for the drywell bypass leakage prior to the termination of fission product release at two hours is reasonable.

3.1.1.2 Aerosol Deposition Within the Drywell

In its evaluation, the staff used a simplified model developed by the staff's contractor for estimating the fission product aerosol deposition by natural processes in the drywell of boiling water reactors (BWRs) following a postulated LOCA. The model is described in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containment." This model was derived by correlating the results of Monte Carlo uncertainty sampling analyses and assessing uncertainties in aerosol properties, drywell geometries, accident progression, and aerosol behavior expected to be associated with a postulated LOCA in the drywell.

The staff assumed that the fission product aerosols in the drywell are removed by natural processes (gravitational sedimentation and phoretic phenomena such as diffusiophoresis and thermophoresis). The staff assumed that the drywell is well mixed during the entire duration of the accident. The aerosol removal rates used by the staff represent the 90th percentile of the uncertainty distributions (see Table 2). For the MSLs, the licensee did not request and the staff has not provided any credit for aerosol deposition.

3.1.2 Containment Leakage Pathway

The primary containment consists of a drywell, a wetwell, and supporting systems to limit fission product leakage during and following the postulated LOCA with rapid isolation of the containment boundary penetrations. The current maximum allowable primary containment leakage rate (L_a), is 0.437 percent of primary containment air weight per day. This value is based on 0.35 percent per day from the containment leak and an additional 100 scfh (0.087 percent per day) through the steam lines. In this amendment request, the licensee proposed (and the staff used in its evaluation) 0.682 percent per day based on 0.385 percent per day from the containment leak and an additional 250 scfh (0.297 percent per day) through the steam lines.

The staff stated in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors," that these leak rates may be reduced 24 hours into the postulated LOCA, if supported by plant configuration and analyses, to a value not less than 50 percent of the TS leak rate limit. The licensee provided the

design-basis drywell and containment pressure profiles for GGNS with adjustments made to the additional hydrogen introduced into the drywell and containment, and the additional heat load associated with increased metal-water reaction and hydrogen ignition. The staff reviewed the licensee's submittal and accepted the 50 percent reduction in the drywell and containment leak rate after 24 hours.

The GGNS secondary containment (the Enclosure Building which surrounds the primary containment) will collect and retain any fission product leakage from the primary containment and will release fission products to the environment through the SGTS following the postulated LOCA. During normal plant operation, the containment is maintained at a slight negative pressure at a vacuum of 0.4-inch water gauge. The GGNS UFSAR states that the secondary containment pressure is expected to remain negative following the postulated LOCA. However, for a short period (three minutes), it may not be maintained below the negative pressure of 0.25-inch water gauge. Therefore, the licensee assumed, and the staff agrees, that the entire primary containment leakage is released directly to the environment during the first three minutes of the postulated LOCA. After three minutes, the SGTS draws 4000 cfm of secondary containment atmosphere air through a HEPA filter with a 99 percent aerosol removal efficiency and a charcoal adsorber with a 99 percent iodine removal efficiency before release to the environment.

3.1.2.1 SGTS

The SGTS is an ESF system and is designed to collect, process, and release the fission product leakage from the primary containment into the Enclosure Building. The SGTS is a redundant system consisting of two 100 percent capacity subsystems. Each subsystem has a design capacity of 4000 cfm and consists of, among other things, a pre-HEPA filter, a 4-inch deep charcoal adsorber, and a post-HEPA filter. The system is designed to Seismic Category 1 standards and is located in a Seismic Category 1 structure.

3.1.2.2 Containment Spray

The containment spray system (CSS) is an ESF system and is designed to provide containment cooling and fission product removal in the containment following the postulated LOCA. The CSS consists of two redundant and independent loops. Each loop has a design spray water flow capacity of 5650 gallons per minute (gpm). The system is designed to Seismic Category 1 standards and is located in a Seismic Category 1 structure. No chemical additives are used in the CSS (see Section 3.1.4).

Before the CSS is activated (30 minutes after accident initiation), the licensee assumed that a mixing rate of two unsprayed volumes per hour between the sprayed and unsprayed portions of the containment atmosphere consistent with the guidance provided in Standard Review Plan (SRP) Section 6.5.2. During the CSS operation, the licensee assumed a mixing rate of 7.5 unsprayed volumes per hour between the sprayed and unsprayed portions of the containment atmosphere with 70,000 cfm of dry air flow. The proposed mixing rate is higher than that specified in SRP Section 6.5.2. The staff accepted the proposed higher mixing rate during the CSS operation on the basis of its review of the licensee's calculation that demonstrated that an adequate mixing flow existed between unsprayed and sprayed regions by natural convection. The spray condenses steam in the containment atmosphere and the movement of the condensation creates additional mixing of the containment air.

In its evaluation, the staff used a simplified model for estimating the fission product aerosol removal by containment sprays following a postulated LOCA. The model is described in NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays." This model was derived by correlating the results of Monte Carlo uncertainty sampling analyses assessing the uncertainties in aerosol properties, aerosol behavior, spray droplet behavior, and the initial and boundary conditions expected to be associated with a postulated LOCA in the containment. Two parameters used in this evaluation are not treated as uncertainty distributions for GGNS: spray water flux and spray fall height. These parameters are plant specific and their values are listed in Table 3. The staff used 90th percentile uncertainty distributions for fission product in aerosol form in calculating the radiological consequences. The major parameters used, including spray removal rates for elemental iodine in the sprayed region, are listed in Table 3.

3.1.3 Post-LOCA Leakage Pathway From ESFs Outside Containment

Any leakage water from ESF components located outside the primary containment releases fission products during the recirculating phase of long-term core cooling following a postulated LOCA. The licensee calculated this leakage to be less than 1.02 gpm and assumed ESF leakage to begin 10 minutes into the postulated LOCA through the entire duration of the accident (30 days). The staff finds the leakage value calculated by the licensee and the leakage initiation time to be reasonable. The licensee also assumed that 10 percent of all forms of iodine contained in the leakage is released directly to the environment consistent with Regulatory Guide 1.183.

3.1.4 Post-Accident Containment Water Chemistry Management

In NUREG-1465, the staff concluded that iodine entering the containment from the reactor coolant system during an accident would be composed of at least 95 percent cesium iodide, with no more than five percent of iodine and hydriodic acid. The licensee conservatively assumed that all five percent of this release is in the form of hydriodic acid in order to maximize the acid generation. Once in the containment, highly soluble cesium iodide will readily dissolve in water pools forming iodide in solution. The radiation-induced conversion of iodide in water into elemental iodine is strongly dependent on the pH. The staff stated in the NUREG that without pH control, a large fraction of iodine dissolved in water pools in ionic form will be converted to elemental iodine and will be reevolved into the containment atmosphere if the pH is less than seven. On the other hand, if the pH is maintained above seven, very little (less than one percent) of the dissolved iodine will be converted to elemental iodine.

The licensee developed and submitted the methodologies used to calculate the post-accident suppression pool water pH transient for determining long-term iodine reevolution from the pool water into the containment atmosphere. In addition, the licensee provided a complete post-accident suppression pool water pH analysis. The staff finds that the licensee followed closely the models and methodologies described in NUREG/CR-5950, "Iodine Evolution and pH Control," NUREG/CR-5732, "Iodine Chemical Forms in LWR [Light Water Reactor] Severe Accidents," and NUREG-1465 for determining (1) the formation of hydrochloric and nitric acids, (2) the suppression pool pH transient, and (3) long-term iodine reevolution. The staff previously used and has accepted the methods in NUREG/CR-5950 and NUREG/CR-5732. The staff finds the licensee's methods used to calculate the post-accident suppression pool water pH transient for determining long-term iodine reevolution to be acceptable.

Using its methodologies, the licensee has determined that the suppression pool water pH rises steadily during fuel gap and in-vessel releases from the initial pH value of 5.3 to above 7.0 due to the introduction of cesium hydroxide into the pool. The pH then begins to decrease after the in-vessel release terminates due to the continued formation of nitric acid in the suppression pool and the formation of hydrochloric acid from radiolytic decomposition of the electrical cable jacketing. After approximately four days, the suppression pool water pH value decreases to less than 7.0.

The staff believes that, for the first 24 hours into the postulated LOCA, the fission product source term behavior, its transport, and release to the environment will be entirely dominated by thermal hydraulic conditions in the drywell and in the containment (drywell leakage, steam production and condensation, and mixing), and by aerosol removal mechanisms (containment spray and aerosol deposition) independent of suppression pool water pH and iodine reevolution from the suppression pool to the containment atmosphere. Consequently, any postulated radiological consequences at any point on the EAB for a 24 hour period will not be affected by iodine reevolution and pH control.

The licensee stated that, in the event of an unmitigated LOCA such as the staff postulated to occur, the GGNS Severe Accident Procedures direct the plant operator to inject the standby liquid control system (SLCS) solution into the reactor vessel in the early stages of the accident for both vessel inventory and re-criticality protection when the core is re-flooded. The SLCS is a safety-related system and designed to Seismic Category 1 standards. It is designed as a reactivity control system and provides backup capability so as to be able to shut down the reactor if the normal control becomes inoperable. The GGNS TS requires the system to be maintained in an operable status whenever the reactor is critical. The system is manually initiated from the main control room to add a boron neutron absorber solution (sodium pentaborate) to the reactor vessel. The SLCS contains at least 5800 pounds of sodium pentaborate. Sodium pentaborate dissolves in water, producing boric acid and sodium borate:

$$Na_2B_{10}O_{16} + 15\ H_2O \rightarrow 2\ Na^+ + 2\ H_2BO_3^- + 8\ H_3BO_3$$

Since boric acid is a relatively weak acid and sodium hydroxide is a strong base, their solution has a buffering effect and will maintain the pH of the suppression pool water at pH values higher than 7.0. The sodium pentaborate solution will be well mixed with the suppression pool water by the end of a 24 hour period as a result of reflooding the reactor vessel. The licensee stated that as a backup, 5000 pounds each of anhydrous borax and boric acid would be mixed in the condensate storage tank in accordance with GGNS Emergency Procedure. This will produce, approximately, an additional 10,000 pounds of sodium pentaborate. The licensee stated, and the staff agrees, that sodium pentaborate from the SLCS is capable of controlling and maintaining long-term suppression pool water pH levels at 7.0 or above from the first 24 hours through the entire 30-day period of the postulated accident.

3.1.5 Control Room Habitability

The GGNS control room is normally maintained at a slightly positive pressure to prevent the introduction of air into the control room from sources other than the 2000 cfm outdoor air makeup flow. The licensee proposed to manually isolate the control room air intakes no later than 20 minutes after the initiation of the postulated LOCA. The normal control room air intake rate of 2000 cfm (without filtration through the CRFA system) is assumed for the first

20 minutes, with an additional 10 cfm assumed to account for control room ingress/egress. Once the air intakes are isolated, the control room atmosphere is recirculated through the control room AC units and 4000 cfm of this flow is processed through the CRFA system. The CRFA system is a redundant system. Each subsystem consists of, among other things, a pre-filter, a HEPA filter, a charcoal adsorber, and a post-HEPA filter. The staff assumed a removal efficiency of 99 percent for fission products in particulate form for the HEPA filter. The licensee has not requested and the staff has not provided any iodine removal credit for the charcoal absorber. After three days, the CRFA system draws outside air at 4000 cfm through HEPA filters into the control room envelope and the recirculation flow terminates. The staff assumed an unfiltered air inleakage rate of 2010 cfm (consisting of 2000 cfm allowable leakage plus an assumed 10 cfm to account for control room ingress / egress) into the control room during the entire 30-day accident period while the control room is isolated.

The licensee reevaluated the control room habitability with the application of the AST and concluded that the radiological consequences to the control room operator resulting from the postulated LOCA are within the 5 rem total effective dose equivalent (TEDE) criterion specified in 10 CFR 50.67. The licensee reached this conclusion:

(1) using the revised atmospheric relative concentrations at the control room air intake,
(2) with manual isolation of control room at 20 minutes after the initiation of the postulated LOCA,
(3) with an unfiltered air inleakage of up to 2010 cfm (2000 cfm allowable leakage plus an assumed 10 cfm to account for control room ingress / egress) into the control room during the entire period of the accident while the control room is isolated, and
(4) taking no credit for iodine removal by the CRFA system.

To verify the licensee's radiological consequence assessments, the staff performed confirmatory radiological consequence dose calculations for the control room operator. The results are given in Table 1, alongside the licensee's results. The staff calculated a TEDE of 3.7 rem to the control room operator from the postulated LOCA. This is within the 5 rem TEDE dose criterion specified in 10 CFR 50.67. In addition, the staff's sensitivity study indicates that the GGNS control room can take in approximately 3500 cfm of unfiltered air inleakge and still meet the relevant dose criterion.

Therefore, the staff concludes that adequate radiation protection is provided to the control room operator to permit access to, and occupation of, the control room under accident conditions without personnel receiving radiation exposures exceeding the 5 rem TEDE dose criterion specified in 10 CFR 50.67. Therefore, the staff finds that the control room habitability assessment performed by the licensee is acceptable.

3.1.6 Resulting Radiological Consequences from the Postulated LOCA

The staff calculated a TEDE of 9.0 rem at the EAB, a TEDE of 3.8 rem at the LPZ boundary, and a TEDE of 3.7 rem in the control room resulting from the postulated LOCA. As shown in Table 1, these calculated dose values are all within the dose criteria specified in 10 CFR 50.67. Therefore, the staff concludes that the requested license amendment, considering the radiological consequences of the postulated LOCA, is acceptable.

3.2 Fuel Handling Accident

The current radiological consequence analysis of the design basis FHA using the TID-14844 source term is provided in the GGNS UFSAR Section 15.7.4. In this license amendment request, the licensee reanalyzed the FHA with the implementation of the AST and presented its results in Attachment 6 of the January 21, 2000, submittal.

In GGNS License Amendment No. 139, issued on October 20, 1999, "Operational Conditions for Handling Irradiated Fuel in the Primary and Secondary Containments," the licensee defined the term "recently irradiated fuel" as irradiated fuel assemblies that contain sufficient fission products to require the operability of ESF systems to meet the relevant offsite and control room operator dose criteria. This term is a plant-specific parameter and is evaluated for each fuel cycle by the licensee. In Amendment No. 139, the licensee and the staff determined that, using the TID-14844 source term, an eight-day decay of irradiated fuel assemblies is sufficient to assure that the radiological consequences of a FHA will be within the relevant dose acceptance criteria specified in SRP Section 15.0.1 and General Design Criterion 19.

In this proposed amendment, the licensee requested that this new term be defined as a decay period of 24 hours for the current fuel cycle with the application of the AST. The licensee concluded that this decay period will be sufficient to meet the TEDE dose criterion in SRP Section 15.0.1 and the criterion specified in 10 CFR 50.67. The licensee reached this conclusion assuming that:

(1) after 24 hours of fission product decay, irradiated fuel assemblies are moved without secondary containment integrity,
(2) core alterations are performed without secondary containment integrity,
(3) the control room is not isolated, and
(4) the CRFA system is not available to remove airborne fission products.

To verify the licensee's radiological consequence assessments, the staff performed confirmatory radiological consequence dose calculations for the following most limiting fuel drop scenario that will produce the most radiological consequence. The scenario involves the drop of an irradiated fuel assembly onto the core without secondary containment after 24 hours of fission product decay in the fuel. The results of the staff's independent radiological consequence calculation are provided in Table 1, along with those results calculated and provided by the licensee. The major parameters and assumptions used by the staff in the radiological consequence calculations are listed in Table 5. The staff calculated a TEDE of 1.8 rem at the EAB and a TEDE of 1.9 rem in the control room. The radiological consequences at the EAB and in the control room calculated by the staff and the licensee are well within the dose criterion specified in 10 CFR 50.67 (5 rem TEDE in the control room), and meet the dose acceptance criterion specified in SRP Section 15.0.1 (a 6.3 rem TEDE at the EAB). Therefore, the staff finds the license amendment requested by the licensee to change GGNS TS Sections 3.7.4, 3.8.2, 3.8.5, and 3.8.8 to be acceptable.

3.3 Control Rod Drop Accident

The current radiological consequence analysis of the design-basis CRDA using the TID-14844 source term is provided in GGNS UFSAR Section 15.4.9. In this license amendment request,

the licensee reanalyzed the CRDA with the application of the AST and presented its results in Attachment 4 of the October 26, 2000, supplemental submittal. The staff reviewed the licensee's accident source term, fission product transport, and removal models, the release pathways, and the major assumptions and parameters used to calculate the radiological consequences. The staff finds that they are acceptable.

To verify the licensee's radiological consequence calculations, the staff performed a confirmatory radiological consequence dose calculation. The results are provided in Table 1, along with the licensee's results. The major parameters and assumptions used by the staff in the radiological consequence calculations are listed in Table 6. The staff calculated a TEDE of less than 1 rem each at the EAB, in the LPZ, and in the control room. The radiological consequences calculated by the staff and the licensee are well within the dose criterion specified in 10 CFR 50.67 (5 rem TEDE in the control room), and meet the dose acceptance criterion specified in SRP Section 15.0.1 (6.3 rem TEDE at EAB and in the LPZ). Therefore, the staff finds the license's reanalysis for the CRDA using the AST to be acceptable.

3.4 Atmospheric Relative Concentration Estimates

The licensee used five years of onsite meteorological data collected during calendar years 1995 through 1999 to estimate the atmospheric relative concentration (X/Q) values used in the dose assessments described above. These X/Q values are listed in Tables 5 through 7. Meteorological data were measured at 10 and 50 meters above grade. Joint recovery of wind speed, wind direction, and atmospheric stability data during the entire five year period met the recommended minimum of 90 percent cited in Regulatory Guide 1.23, "Onsite Meteorological Programs," although data recovery was slightly less than this recommended recovery rate during 1995 and 1998. The licensee examined the data and performed year-to-year comparisons to confirm the overall quality of the data.

The licensee used the PAVAN computer code to calculate X/Q values for the EAB and LPZ. This methodology is described in Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants." As input to this calculation, the licensee sorted the five years of meteorological data to approximate a single year of data in joint frequency distribution form. The staff made confirmatory estimates using the five years of data in hourly form. Resultant staff X/Q estimates were somewhat higher for the EAB and LPZ for the 0-2 hour time period, and lower for the LPZ for the longer time periods. The licensee performed additional calculations to demonstrate that the X/Q values used in their dose assessment were bounded by their prior X/Q calculations. The staff has concluded that the difference between the licensee and staff X/Q estimates does not affect the staff's conclusion that the resultant dose estimates in support of this amendment meet regulatory requirements.

The licensee used the ARCON96 methodology to calculate X/Q values for control room dose assessment. ARCON96 is described in NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wake." Estimates were based on the five-year period of meteorological data in hourly form. All postulated releases were calculated as ground level point releases assuming no release flow. Where appropriate, estimates were made from each postulated release location assuming possible intake at either the Safeguard Switchgear and Battery Ventilation Intake or the Control Room Roof Intake. The higher X/Q value for each time period was then used in the dose assessment. The FHA control room X/Q value was

calculated as an effective X/Q by estimating a weighted average of the Containment Vent and SGTS X/Q values. The staff concludes that the licensee's control room X/Q estimates are acceptable.

3.5 Conclusions

The staff has reviewed the licensee's analysis and performed a confirmatory assessment of the radiological consequence of the postulated LOCA, FHA, and CRDA. The doses calculated by the staff and the licensee are listed in Table 1. The doses are all within relevant dose criteria specified in 10 CFR 50.67 and SRP Section 15.0.1. Therefore, the staff concludes that the radiological consequences analyzed and submitted by the licensee are acceptable. On the basis of this evaluation, the staff further concludes that the license amendment requested by the licensee is acceptable.

Table 1
Radiological Consequences Expressed as TEDE
(rem)

DBAs	EAB		LPZ		Control Room	
	NRC	GGNS	NRC	GGNS	NRC	GGNS
LOCA	9.0	8.779	3.8	4.601	3.7	3.647
Dose criteria[1]	25		25		5.0	
FHA	1.8	1.982	NA[2]	NA[2]	1.9	2.035
Dose criteria	6.3[3]				5.0[1]	
CRDA	0.1	0.147	<0.1	0.064	0.26	0.26
Dose criteria	6.3[3]		6.3[3]		5.0[1]	

[1] 10 CFR 50.67
[2] Not Applicable
[3] Dose acceptance criteria in SRP Section 15.0.1

Table 2
Parameters and Assumptions Used in Radiological Consequence Calculations MSIV Leakage Pathway

Parameter	Value
Reactor power (102% design power level)	3910 MWt
Drywell volume	2.7×10^5 ft^3
Leakage rate for intact MSLs	
0 to 24 hours	150 ft^3/hr
1 to 30 days	75 ft^3/hr
Leakage rates for ruptured MSL	
zero to 24 hours	100 ft^3/hr
1 to 30 days	50 ft^3/hr
Release pathways	
0 to 20 minutes	To environment
20 minutes to 30 days	To secondary containment
Aerosol removal rate in drywell (per hour)	
Hours	Rates
0 to 0.5	0.7474
0.5 to 2.0	0.2983
2.0 to 5.0	1.0550
5.0 to 8.3	0.6390
8.3 to 12	0.5571
12 to 19.4	0.5236
19.4 to 24	0.5068
Aerosol removal rate in main steam lines	0
Elemental iodine removal rate in drywell	0.866 per hour

Table 3
Parameters and Assumptions
used in
Radiological Consequence Calculations
Containment Leakage Pathway

Parameter	Value
Reactor power	3910 MWt
Volume of sprayed region	$8.4x10^5$ ft^3
Volume of unsprayed region	$5.6x10^5$ ft^3
Flow rate from drywell to unsprayed region	
0 - 2 hours	3000 ft^3/min
2 hours - 30 days	Well mixed
Flow rate from unsprayed region to drywell	
0 - 2 hours	0 ft^3/min
2 hours - 30 days	Well mixed
Flow rate between drywell and sprayed region	0 ft^3/min
Flow rate from sprayed region to unsprayed region	
0 - 0.5 hours	18,667 ft^3/min
0.5 - 24 hours	70,000 ft^3/min
Flow rate from unsprayed region to sprayed region	
0 - 0.5 hours	18,667 ft^3/min
0.5 - 24 hours	70,000 ft^3/min
Containment leak rate to environment	
0 - 24 hours	0.682% per day
1 - 30 days	0.341% per day
Spray removal rates	
Aerosols	9.51 per hour
Time to reach DF[1] of 50	3 hours
Elemental iodine	20 per hour
Time to reach DF[1] of 200	6.1 hours
Spray water flux	0.03177 (cm)/(cm sec)
Spray fall height	64 ft

[1] Decontamination factor

Table 4
Parameters and Assumptions
Used in
Radiological Consequence Calculations
ECCS Leakage Pathway

ECCS Leakage Model

Parameter	Value
Plant power	3910 MWt
Release fractions and timing	As specified for BWR in NUREG-1465 (gap and early in-vessel iodine releases only)
Release location	Directly to suppression pool
Suppression pool water volume	1.706×10^5 ft^3
Secondary containment volume	3×10^5 ft^3
ECCS leak rate	
10 minutes to 30 days	0.15 ft^3/min
Partition factor	10
SGTS	
Flow rate	4000 cfm
Iodine removal efficiency	99%
Aerosol removal efficiency	99%

Table 5
Parameters and Assumptions
Used in
Radiological Consequence Calculations
FHA

Parameter	Value
Reactor power	3910 MWt
Radial peaking factor	1.7
Fission product decay period	24 hours
Number of fuel rods damaged	142
Fuel pool water depth	23 ft
Fuel gap fission product inventory	
Noble gases excluding Kr-85	5 percent
Kr-85	10 percent
I-131	8 percent
Alkali metals	12 percent
Fuel pool decontamination factors	
Iodine	200
Noble gases	1
Secondary containment requirements	None
Control room	
Isolation	No
Fresh air system	No credit
Unfiltered infiltration	Not applicable
Recirculation flow	Not applicable
Iodine Protection factor	1
Atmospheric relative concentrations (χ/Q values)	
EAB (0 to 2 hours)	6.0E-4 sec/m^3
Control room	8.5E-4 sec/m^3
Duration of accident	2 hours
Computer code used in dose calculation	Habitt 1.1
Dose conversion factors	Federal Guidance Reports 11 and 12

Table 6
Parameters and Assumptions
Used in
Radiological Consequence Calculations
CRDA

Parameters	Values
Source term	NUREG-1465
Reactor power	3910 MWt
Core radial peaking factor	1.7
Condense airborne volume	2.27E+5 ft^3
Turbine airborne volume	5.65E+4 ft^3
Gap release fractions	
Noble gases	10 percent
Halogens	10 percent
Alkali metals	12 percent
Core melt fractions	
Noble gases	100 percent
Halogens	50 percent
Alkali metals	25 percent
Fraction of fuel melted in each fuel assembly	0.77 percent
Number of fuel assemblies affected	16
Transfer rate from condenser to environment	1 percent per day
Fission product release fractions from each fuel assembly	
Noble gases	10.308 percent
Halogens	10.693 percent
Alkali metals	12.1 percent
Release fractions to turbine/condenser	
Noble gases	100 percent
Iodine	10 percent
Particulate	1 percent
Atmospheric dispersion factors (sec/ m^3)	
0 to 2 hours EAB	6.0E-4
0 to 2 hours LPZ	1.25E-4
2 to 8 hours LPZ	6.0E-5
8 to 24 hours LPZ	4.5E-5

Table 6
(Continued)
Parameters and Assumptions
Used in
Radiological Consequence Calculations
CRDA

Control Room

Volume	2.53E+5 ft^3
Manual isolation	30 minutes
Automatic isolation	No
Fresh air supply system fission product removal efficiencies	
HEPA filter	99 percent
Charcoal adsobers	0 percent
Recirculation rate	4,000 cfm (after 18 minutes)
Unfiltered inleakage rate	2010 cfm (from 0 to 30 days)
Control room atmospheric dispersion factors (sec/ m^3)	
0 to 2 hours	8.0E-4
2 to 8 hours	7.0E-4
8 to 24 hours	3.0E-4
Source term release point	Turbine building vent
Air intake point into control room	Control building roof

Table 7

Meteorological Data
for
LOCA

EAB

Time (hr)	X/Q (sec/m^3)
0-2	6.0E-4

LPZ

Time (hr)	X/Q (sec/m^3)
0-2	1.25E-4
2-8	6.0E-5
8-24	4.5E-5
24-96	2.0E-5
96-720	7.0E-6

Control Room

Time (hr)	X/Q (sec/m^3)
0-2	8.0E-4
2-8	5.0E-4
8-24	2.5E-4
24-96	1.6E-4
96-720	1.3E-4

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Mississippi State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (65 FR 15380, dated March 22, 2000). The October 26, 2000 and February 22, 2001, supplemental letters contained changes to the TS pages and to the operating license page submitted with the original application. The June 29, September 1, and December 22, 2000, supplemental letters provided clarifying information regarding the original application. These changes and clarifying information do not affect NRC's finding of no significant hazards consideration determination. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: J. Y. Lee
K. Parczewski
L. Brown

Date: March 14, 2001

The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice. The changes shall be implemented within 60 days from the date of issuance. You are requested to inform the staff by letter when these changes have been implemented at GGNS. If you have any questions regarding this subject, please contact me at 301-415-2623.

Sincerely,

/RA/

S. Patrick Sekerak, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-416

Enclosures: 1. Amendment No. 145 to NPF-29
2. Safety Evaluation

cc w/encls: See next page

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