September 23, 1986

Docket No.: 50-416

DISTRIBUTION:

Docket File	Young, OGC	EButcher
NRC PDR	RDiggs, LFMB	NThompson
Local PDR	TBarnhart (4)	JPartlow
PD#4 Rdg File	EJordan	
RBernero	LHarmon	
WButler	CGrimes	
LKintner	WJones	
MO'Brien	DVassallo	
	ACRS (10)	
	CMiles, OPA	

Mr. Oliver D. Kingsley, Jr.
Vice President, Nuclear Operations
Mississippi Power & Light Company
Post Office Box 23054
Jackson, Mississippi 39205

Dear Mr. Kingsley:

SUBJECT: CHANGES TO TECHNICAL SPECIFICATIONS TO ADD A TRANSFER SWITCH AND MAKE ADMINISTRATIVE CHANGES TO CORRECT ERRORS

RE: GRAND GULF NUCLEAR STATION, UNIT 1

The Commission has issued the enclosed Amendment No.19 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your applications dated April 14, 1986 and May 12, 1986.

This amendment adds a transfer switch to the list of remote shutdown system controls and makes other administrative changes to correct errors. Changes on Technical Specification Pages 3/4 3-18, 3/4 3-88, B 3/4 3-2, 5-2 and 5-6 are effective upon issuance of this amendment. Changes on Technical Specification Page 3/4 3-71 are effective when the equipment necessitating the changes on that page is installed and operable. For those changes that are not effective upon issuance, you are requested to inform the NRR by letter of their effective dates within 7 days of the date the equipment is made operable.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by:

Lester L. Kintner, Project Manager
BWR Project Directorate No. 4
Division of BWR Licensing

Enclosures:
1. Amendment No. 19 to License No. NPF-29
2. Safety Evaluation

cc w/enclosures:
See next page

Previously concurred*:

PD#4/LA*	PD#4/PM*	OGC*	PD#4/D*
MO'Brien	LKintner:lb	Young	WButler
09/10/86	09/10/86	09/18/86	09/23/86

cc:
Robert B. McGehee, Esquire
Wise, Carter, Child, Steen and Caraway
P.O. Box 651
Jackson, Mississippi 39205

Nicholas S. Reynolds, Esquire
Bishop, Liberman, Cook, Purcell
and Reynolds
1200 17th Street, N.W.
Washington, D. C. 20036

Mr. Ralph T. Lally
Manager of Quality Assurance
Middle South Services, Inc.
P.O. Box 61000
New Orleans, Louisiana 70161

Mr. Larry F. Dale, Director
Nuclear Licensing and Safety
Mississippi Power & Light Company
P.O. Box 23054
Jackson, Mississippi 39205

Mr. R. W. Jackson, Project Engineer
Bechtel Power Corporation
15740 Shady Grove Road
Gaithersburg, Maryland 20877-1454

Mr. Ross C. Butcher
Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Route 2, Box 399
Port Gibson, Mississippi 39150

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission,
101 Marietta Street, N.W., Suite 2900
Atlanta, Georgia 30323

Mr. J. E. Cross
Grand Gulf Nuclear Station Site Director
Mississippi Power & Light Company
P.O. Box 756
Port Gibson, Mississippi 39150

Mr. C. R. Hutchinson
GGNS General Manager
Mississippi Power & Light Company
Post Office Box 756
Port Gibson, Mississippi 39150

The Honorable William J. Guste, Jr.
Attorney General
Department of Justice
State of Louisiana
Baton Rouge, Louisiana 70804

Office of the Governor
State of Mississippi
Jackson, Mississippi 39201

Attorney General
Gartin Building
Jackson, Mississippi 39205

Mr. Jack McMillan, Director
Division of Solid Waste Management
Mississippi Department of Natural
Resources
Bureau of Pollution Control
Post Office Box 10385
Jackson, Mississippi 39209

Alton B. Cobb, M.D.
State Health Officer
State Board of Health
P.O. Box 1700
Jackson, Mississippi 39205

President
Claiborne County Board of Supervisors
Port Gibson, Mississippi 39150

Mr. Ted H. Cloninger
Vice President, Nuclear Engineering
and Support
Mississippi Power & Light Company
Post Office Box 23054
Jackson, Mississippi 39205

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

MISSISSIPPI POWER & LIGHT COMPANY
MIDDLE SOUTH ENERGY, INC.
SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION
DOCKET NO. 50-416
GRAND GULF NUCLEAR STATION, UNIT 1
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 19
License No. NPF-29

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

   A. The applications for amendment by Mississippi Power & Light Company, Middle South Energy, Inc., and South Mississippi Electric Power Association, (the licensees) dated April 14, 1986 and May 12, 1986, comply 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:

   Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 19 , are hereby incorporated into this license. Mississippi Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. Technical Specification Pages 3/4 3-18, 3/4 3-88, B 3/4 3-2, 5-2 and 5-6 in this amendment are effective upon issuance of this amendment. The change made on Technical Specification Page 3/4 3-71 is effective when the equipment necessitating the change on this page is installed and made operable.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:

Walter R. Butler, Director
BWR Project Directorate No. 4
Division of BWR Licensing

Attachment: Changes to the Technical Specifications

Date of Issuance: September 23, 1986

Previously concurred*:

PD#4/LA*	PD#4/PM*	OGC*	PD#4/D
MO'Brien	LKintner:lb	MYoung	WButler WB
09/10/86	09/10/86	09/11/86	9/23/86

ATTACHMENT TO LICENSE AMENDMENT NO. 19

FACILITY OPERATING LICENSE NO. NPF-29

DOCKET NO. 50-416

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Overleaf page(s) provided to maintain document completeness.*

Remove	Insert
3/4 3-17	3/4 3-17*
3/4 3-18	3/4 3-18
3/4 3-71	3/4 3-71
3/4 3-72	3/4 3-72*
3/4 3-87	3/4 3-87*
3/4 3-88	3/4 3-88
B 3/4 3-1	B 3/4 3-1*
B 3/4 3-2	B 3/4 3-2
5-1	5-1*
5-2	5-2
-	5-2a
5-5	5-5*
5-6	5-6

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
2. MAIN STEAM LINE ISOLATION (Continued)		
g. Main Steam Line Tunnel Δ Temp. - High	≤ 101°F**	≤ 104°F**
h. Manual Initiation	NA	NA
3. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -41.6 inches*	≥ -43.8 inches
b. Drywell Pressure - High	≤ 1.23 psig	≤ 1.43 psig
c. Fuel Handling Area Ventilation Exhaust Radition - High High	≤ 3.6 mR/hr**	≤ 4.0 mR/hr**
d. Fuel Handling Area Pool Sweep Exhaust Radiation - High High	≤ 30 mR/hr**	≤ 35 mR/hr**
e. Manual Initiation	NA	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High	≤ 79 gpm	≤ 89** gpm
b. Δ Flow Timer	≤ 45 seconds	≤ 57 seconds
c. Equipment Area Temperature - High		
1. RWCU Hx Room	≤ 120°F	≤ 126°F
2. RWCU Pump Rooms	≤ 170°F	≤ 176°F
3. RWCU Valve Nest Room	≤ 135°F	≤ 141°F
d. Equipment Area Δ Temp. - High		
1. RWCU Hx Room	≤ 65°F	≤ 66°F
2. RWCU Pump Rooms	≤ 115°F	≤ 118°F
3. RWCU Valve Nest Room	≤ 70°F	≤ 73°F
e. Reactor Vessel Water Level - Low Low, Level 2	≥ -41.6 inches*	≥ -43.8 inches

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
4. REACTOR WATER CLEANUP SYSTEM ISOLATION (Continued)		
f. Main Steam Line Tunnel Ambient Temperature - High	$\leq$ 185°F**	$\leq$ 191°F**
g. Main Steam Line Tunnel Δ Temp. - High	$\leq$ 101°F**	$\leq$ 104°F**
h. SLCS Initiation	NA	NA
i. Manual Initiation	NA	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a. RCIC Steam Line Flow - High		
1. Pressure	< 56" H_2O	< 64" H_2O
2. Time Delay	5±2 seconds	5±2 seconds
b. RCIC Steam Supply Pressure - Low	$\geq$ 60 psig	$\geq$ 53 psig
c. RCIC Turbine Exhaust Diaphragm Pressure - High	$\leq$ 10 psig	$\leq$ 20 psig
d. RCIC Equipment Room Ambient Temperature - High	$\leq$ 185°F**	$\leq$ 191°F**
e. RCIC Equipment Room Δ Temp. - High	$\leq$ 125°F**	$\leq$ 128°F**
f. Main Steam Line Tunnel Ambient Temperature - High	$\leq$ 185°F**	$\leq$ 191°F**
g. Main Steam Line Tunnel Δ Temp. - High	$\leq$ 101°F**	$\leq$ 104°F**
h. Main Steam Line Tunnel Temperature Timer	$\leq$ 30 minutes	$\leq$ 30 minutes
i. RHR Equipment Room Ambient Temperature - High	$\leq$ 165°F**	$\leq$ 171°F**
j. RHR Equipment Room Δ Temperature - High	$\leq$ 99°F**	$\leq$ 102°F**
k. RHR/RCIC Steam Line Flow - High	$\leq$ 145" H_2O	$\leq$ 151" H_2O

TABLE 3.3.7.4-1 (Continued)

REMOTE SHUTDOWN SYSTEM CONTROLS

	CONTROL	MINIMUM CHANNELS OPERABLE Div 1	Div 2
12.	RHR Injection Valves	2^{b}	2^{b}
13.	RHR Test Line Valve	1	1
14.	RHR HX Cond. to RCIC Valve	1	1
15.	RHR HX Flow to Suppression Pool Valve	1	1
16.	RHR Discharge to Radwaste Valve	1	1
17.	RCIC Steam to RHR HX Valve	2^{b}	2^{b}
18.	Diesel Generator HX Inlet Valve	1	1
19.	Safety/Relief Valves	6^{b}	6^{b}
20.	Control Room to Shutdown Panel Transfer Switch	1	NA
21.	RCIC Turbine Flow Controller	1	NA
22.	RCIC Suction Flow Suppression Pool Valve	1	NA
23.	RCIC Injection Shutoff Valve	1	NA
24.	RCIC Suction From CST	1	NA
25.	RCIC Recirc. Main Flow Bypass Valve	1	NA
26.	RCIC Test RTN to CST IB Valve	1	NA
27.	RCIC Test RTN to CST OB Valve	1	NA
28.	Steam to RCIC Turbine Valve	1	NA
29.	RCIC Turbine Trip & Throttle Valve	1	NA
30.	RCIC Turbine Cooling Water Valve	1	NA
31.	RCIC Turbine Local Control Select Switch	1	NA
32.	RCIC Gland Seal Compressor	1	NA
33.	Shutdown Cooling Isolation Valve Reset Switch	1	1

NOTE: a. 1 per cooling tower fan
b. 1 per valve

TABLE 4.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Vessel Pressure	M	R
2. Reactor Vessel Water Level	M	R
3. Suppression Pool Water Level	M	R
4. Suppression Pool Water Temperature	M	R
5. RHR System Flow	M	R
6. Standby Service Water System Flow	M	R
7. RCIC Turbine Speed	M	R
8. Condensate Storage Tank Level	M	R

TABLE 3.3.7.9-1 (Continued)
FIRE DETECTION INSTRUMENTATION

ROOM	ELEV	ROOM NAME	MINIMUM INSTRUMENTS OPERABLE* HEAT (X/Y)	FLAME(1) (X/Y)	SMOKE(1) (X/Y)
12. Zone 2-17					16/0
1A101	93'	Passage			
1A109	93'	HPCS Pump Rm.			
1A111	93'	Piping Penetration Rm.			
1A114	93'	Fan Coil Area (Partial)			
1A117	93'	Misc. Equip. Area (Partial)			
1A121	103'	East Corridor			
1A122	103	South Corridor (Partial)			
1A123	103'	North Corridor (Partial)			
13. Zone 2-18					20/0
1A201	119'	East Corridor			
1A211	119'	North Corridor (Partial)			
1A215	119'	South Corridor (Partial)			
14. Zone 2-19					13/0
1A314	139'	South Corridor (Partial)			
1A316	139'	North Corridor (Partial)			
1A321	139'	MCC Area			
1A322	139'	Centrifugal Chiller Area			
1A323	139'	SGTS Area			
1A324	139'	HVAC Equip. Area			
1A326	139'	SGTS Area			
15. Zone 2-20					2/0
1A305	139'	Steam Tunnel			
16. Zone 2-21					4/0
1A12	185'	Stairwell			
1A12	208'	Stairwell			
1A12	245'	Stairwell			
d. DIESEL GENERATOR BUILDING					
1. Zone 2-10			9/0		
1D301	133'	Corridor	0/3 (Deluge)		
1D304	133'	Day Tank Area			
1D306	133'	Div. III Diesel Gen. Room			
1D401	158'	Div. III Diesel Gen. Room	0/7 (Deluge)		
2. 2-11			6/0		
1D303	133'	Day Tank Area			
1D308	133'	Div. II Diesel Gen. Room			
1D402	158'	Div. II Diesel Gen. Room	0/7 (Deluge)		

TABLE 3.3.7.9-1 (Continued)
FIRE DETECTION INSTRUMENTATION

	ROOM	ELEV	ROOM NAME	MINIMUM INSTRUMENTS OPERABLE* HEAT (X/Y)	FLAME(1) (X/Y)	SMOKE(1) (X/Y)
3.	Zone 2-12				6/0	
	1D302	133'	Day Tank Area			
	1D310	133'	Div. I Diesel Gen. Room			
	1D403	158'	Div. I Diesel Gen. Room	0/7 (Deluge)		
e.	STANDBY SERVICE WATER PUMP HOUSE					
1.	Zone 2-1					4/0
	1M110	133'	SSW Pump Rm. A			
	1M112	133'	SSW Valve Rm. A			
	2M110	133'	SSW Pump Rm. B			
	2M112	133'	SSW Valve Rm. B			
f.	CHARCOAL FILTER TRAINS					
1.	Standby Gas Treatment System Filter Trains A & B			2/0 (Allison Thermistor Wire)		
	Auxiliary Building El. 139'					
2.	Control Room Standby Fresh Air System Filter Trains A & B			2/0 (Allison Thermistor Wire)		
	Control Building El. 133'					
g.	CONTROL BUILDING (PGCC HALON SYSTEMS)					
	0C503	166'	Control Room (Unit 1 side)			
			Module/Halon Panel			
			1H13-U700/1H13-P900	0/10		10/0
			1H13-U701/1H13-P901	0/10		15/0
			1H13-U702/1H13-P902	0/9		14/0
			1H13-U703/1H13-P903	0/11		17/0
			1H13-U720/1H13-P920	0/7		13/0
			SH13-U730/SH13-P930	0/11		12/0
			1H13-U738/1H13-P938	0/10		12/0
			SH13-U739/SH13-P939	0/5		14/0

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be absorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the accident analysis. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance. Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. Negative barometric pressure fluctuations are accounted for in the trip setpoints and allowable values specified for drywell pressure-high. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D.C. operated valves, a 3 second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that

ISOLATION ACTUATION INSTRUMENTATION (continued)

the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 10 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 10 second diesel startup. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 10 second delay. It follows that checking the valve speeds and the 10 second time for emergency power establishment will establish the response time for the isolation functions. However, to enhance overall system reliability and to monitor instrument channel response time trends, the isolation actuation instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Negative barometric pressure fluctuations are accounted for in the trip setpoints and allowable values specified for drywell pressure-high. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO-24222, dated December 1979, and Section 15.8 Appendix 15A of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1.2-1.

UNRESTRICTED AREA AND SITE BOUNDARY FOR GASEOUS EFFLUENTS AND FOR LIQUID EFFLUENTS

5.1.3 The UNRESTRICTED AREA AND SITE BOUNDARY for gaseous effluents and for liquid effluents shall be as shown in Figure 5.1.3-1. The gaseous effluent release points are shown in Figure 5.1.1-1.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The containment is a steel lined, reinforced concrete structure composed of a vertical right cylinder and a hemispherical dome. Inside and at the bottom of the containment is a reinforced concrete drywell composed of a vertical right cylinder and a steel head which contains an approximately eighteen to nineteen foot deep water filled suppression pool connected to the drywell through a series of horizontal vents. The containment has a minimum net free air volume of 1,400,000 cubic feet. The drywell has a minimum net free air volume of 270,000 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The containment and drywell are designed and shall be maintained for:

a. Maximum internal pressure:
   1. Drywell 30 psig.
   2. Containment 15 psig.

b. Maximum internal temperature:
   1. Drywell 330°F.
   2. Suppression pool 185°F.

c. Maximum external-to-internal differential pressure:
   1. Drywell 21 psid.
   2. Containment 3 psid.

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of the Auxiliary Building and the Enclosure Building, and has a minimum free volume of 3,640,000 cubic feet.

[THIS PAGE LEFT BLANK INTENTIONALLY]

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 800 fuel assemblies with each fuel assembly containing 62 fuel rods and two water rods clad with Zircaloy -2. Each fuel rod shall have a design nominal active fuel length of 150 inches. The initial core loading shall have a design nominal enrichment of 1.708 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 193 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing a design nominal 143.7 inches of boron carbide, B_4C, powder surrounded by a cruciform shaped stainless steel sheath.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of:

   1. 1250 psig on the suction side of the recirculation pump.

   2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.

   3. 1550 psig from the discharge shutoff valve to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,000 cubic feet at a nominal T_{ave} of 533°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.2-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, including all calculational uncertainties and biases as described in Section 9.1 of the FSAR.

b. A nominal 6.26-inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 202'5 1/4".

CAPACITY

5.6.3 The spent fuel storage capacity is designed and shall be maintained with a storage capacity limited to:

a. No more than 2324* spent fuel assemblies in the spent fuel pool, and

b. No more than 800 spent fuel assemblies in the upper containment pool.

Placement of fuel in the upper containment pool is limited to temporary storage of fuel during refueling operations. Prior to return to reactor criticality, all spent fuel shall be removed from the upper containment pool.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7.1-1.

---

*The physical limit is 4348. The 2324 limit reflects the number of spent fuel assemblies that can be stored in the spent fuel pool without excessive reliance on RHR supplement cooling; i.e., for a time period in excess of a normal refueling duration.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 19 TO FACILITY OPERATING LICENSE NO. NPF-29

MISSISSIPPI POWER & LIGHT COMPANY

MIDDLE SOUTH ENERGY, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By letter dated April 14, 1986, Mississippi Power & Light Company (the licensee) requested an amendment to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. The proposed amendment would make five changes in the Technical Specifications: (1) change Figure 5.1.1-1 "Exclusion Area and Gaseous Effluent Release Points" to show the area for Unit 1 rather than the area for Unit 1 and Unit 2; (2) change the reference for shutdown reactivity calculational uncertainties and biases in Technical Specification 5.6.1.a from Final Safety Analysis Report (FSAR) Section 4.3 to FSAR Section 9.1; (3) change the Technical Specification Bases 3/4.3.2 "Isolation Actuation Instrumentation" to reflect a diesel generator start time that is consistent with the associated Technical Specification; (4) change a Halon panel number in Technical Specification Table 3.3.7.9-1 "Fire Detection Instrumentation" from 1H13-P930 to SH13P-930; and (5) add "control room to shutdown panel transfer switch" to Technical Specification Table 3.3.7.4-1 "Remote Shutdown System Controls."

By letter dated May 12, 1986, the licensee requested an amendment to the operating license which would change Technical Specification Table 3.3.2-2 "Isolation Actuation Instrumentation Setpoints" by changing the setpoint and allowable value for the reactor core isolation cooling (RCIC) system steam line high flow trip.

These two requests are addressed separately in the evaluation below.

2.0 EVALUATION

2.1 April 14, 1986 Request

Figure 5.1.1-1 of the Technical Specifications is intended to identify the exclusion area and gaseous effluent release points for Grand Gulf Nuclear Station (GGNS), Unit 1. As currently depicted in Figure 5.1.1-1, the northern half of the exclusion area for Grand Gulf Unit 1 is centered on the Unit 2 reactor, and the southern half of the exclusion area is centered on the Unit 1 reactor. The minimum exclusion area boundary distance is 696 meters. The licensee has proposed to replace the current envelope of the exclusion areas for both reactors with an exclusion area for Grand Gulf Unit 1 centered solely on the Unit 1 reactor with a radius of 696 meters. For operation of Unit 1 only, the proposed change will not affect the results and conclusions presented in Section 11.2.2 Gaseous Waste Treatment System and Section 15.3, Radiological Consequences of Design Basis Accidents, of our Safety Evaluation Report (NUREG-0831, dated September, 1986) which assumed a minimum exclusion area boundary distance for the Technical Specifications for GGNS Unit 1, since Unit 2 is not in operation.

Technical Specification 5.6.1a concerning spend fuel storage racks presently references Section 4.3 of the Final Safety Analysis Report (FSAR) for a description of uncertainties and biases in the calculation of shutdown reactivity. The licensee has proposed to change the reference to FSAR Section 9.1. Section 9.1 is the appropriate reference because it describes the analysis assumptions used in the nuclear design of the fuel storage facilities whereas Section 4.3 describes analysis assumptions used in the reactor core design. Accordingly, the staff concludes that the change is acceptable.

Bases 3/4.3.2 "Isolation System Actuation Instrumentation" presently uses a diesel generator start time of 13 seconds in the description of the isolation system response time. The proposed change would make the diesel generator start time 10 seconds to make the Bases section consistent with the isolation system instrumentation response times in Table 3.3.2-3 of the Technical Specifications. The times in Table 3.3.2-3 were previously reviewed and approved by the NRC staff in our review of Amendment 13 to the low-power license (NPF-13), transmitted to the licensee by letter dated August 3, 1984. Since the basis for the proposed change has been previously reviewed, we conclude that the change is administrative in nature and is therefore acceptable.

Table 3.3.7.9-1 of the Technical Specification presently includes an incorrect designation (1H13-P930) for one of the fire protection cabinets in the GGNS Unit 1 control room based on the licensee's review of appropriate drawings. The proposed change is to use the correct identification number of the module/Halon Panel in Table 3.3.7.9-1 (SH13-P930). Therefore, the change is acceptable.

Table 3.3.7.4-1 of the Technical Specifications lists the controls on the remote shutdown panel that are required to be operable during startup and power operation. The proposed change would add a control room-to-shutdown panel transfer switch to this table because of a design change being made to fulfill License Condition 2.C.(22). The license condition requires the licensee to install electrical isolation switches between the control room and the Division I remote shutdown panel, such that a fire in the control room will not affect the control capability from a remote shutdown panel. The remote shutdown system consists of two panels, one for Division I and the other for Division II. Since the Division I panel and the Division II panel are located in two separate fire zones, a fire in the Division I remote shutdown panel area will not affect Division II remote shutdown circuits. Therefore, only circuits for one division are required to be isolated from the room. The proposed design consists of a "master" switch and approximately 36 lockout relays. The operator only has to throw one switch, to remove the control functions from all Division I control room circuits that are duplicated on the Divison I remote shutdown panel. The Division II remote shutdown circuits are not isolated from the control room. The design also gives the operator the capability of selective restoration of control functions to the control room from the Division I remote shutdown panel. These design changes were approved by staff's letter dated February 15, 1985. Accordingly, the staff finds that the changes to Table 3.3.7.4-1 of the Technical Specifications are acceptable.

2.2 <u>May 12, 1986 Request</u>

Table 3.3.2-2 of the Technical Specifications lists the trip setpoints and allowable values for isolation actuation instrumentation. For Item 5.a.1 of this table, the reactor core isolation cooling (RCIC) steam line high flow trip, the present trip setpoint (363 inches of water) and allowable value (373 inches of water) were found by the licensee to be incorrect during its review of the results of RCIC startup tests. The proposed changes to Table 3.3.2-2 would provide the correct trip setpoint (56 inches of water) and the correct allowable value (0.64 inches of water).

The licensee's review of the startup test data revealed that lower than anticipated differential pressures were being recorded, which prompted the evaluation of the setpoint for RCIC system steam line flow. The discrepancy was due to the use of a larger RCIC pump and larger piping for the Grand Gulf design compared to the generic design. The generic RCIC system uses a 500 gpm pump and a 4" diameter RCIC turbine steam supply line, while the Grand Gulf's RCIC system uses an 800 gpm pump and a 6" diameter RCIC turbine steam supply line. Use of the generic RCIC system parameters in the analysis of the Grand Gulf design resulted in a significantly higher calculated steam velocity following a postulated RCIC loss of coolant accident (LOCA) than would be experienced with the Grand Gulf design, and hence a higher calculated differential pressure across the flow element.

After the licensee identified the discrepancy, the NSSS vendor (General Electric) modified the computational model to incorporate the 6" line and calculated the revised analytical limit of 73.3 inches of water. This is the differential pressure corresponding to a flow rate of 300% of rated RCIC steam flow, which was the flowrate assumed in the safety analysis to RCIC LOCA. The revised allowable value of 64 inch H_2O includes allowance for combined instrument accuracy and calibration errors. The revised trip setpoint of 56 inches of water includes allowance for instrument drift.

The proposed change does not impact the results the GGNS safety analysis for RCIC LOCA because the flow used in the safety analysis was 300% of rated RCIC steam flow. The revised values for Table 3.3.2-2 are the correct values for the differential pressure corresponding to a flow rate of 300% of rated RCIC steam flow for the GGNS plant. Accordingly, the staff finds that the changes to Table 3.3.2-2 of the Technical Specifications are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The 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 proposed findings that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this 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 this amendment.

4.0 CONCLUSION

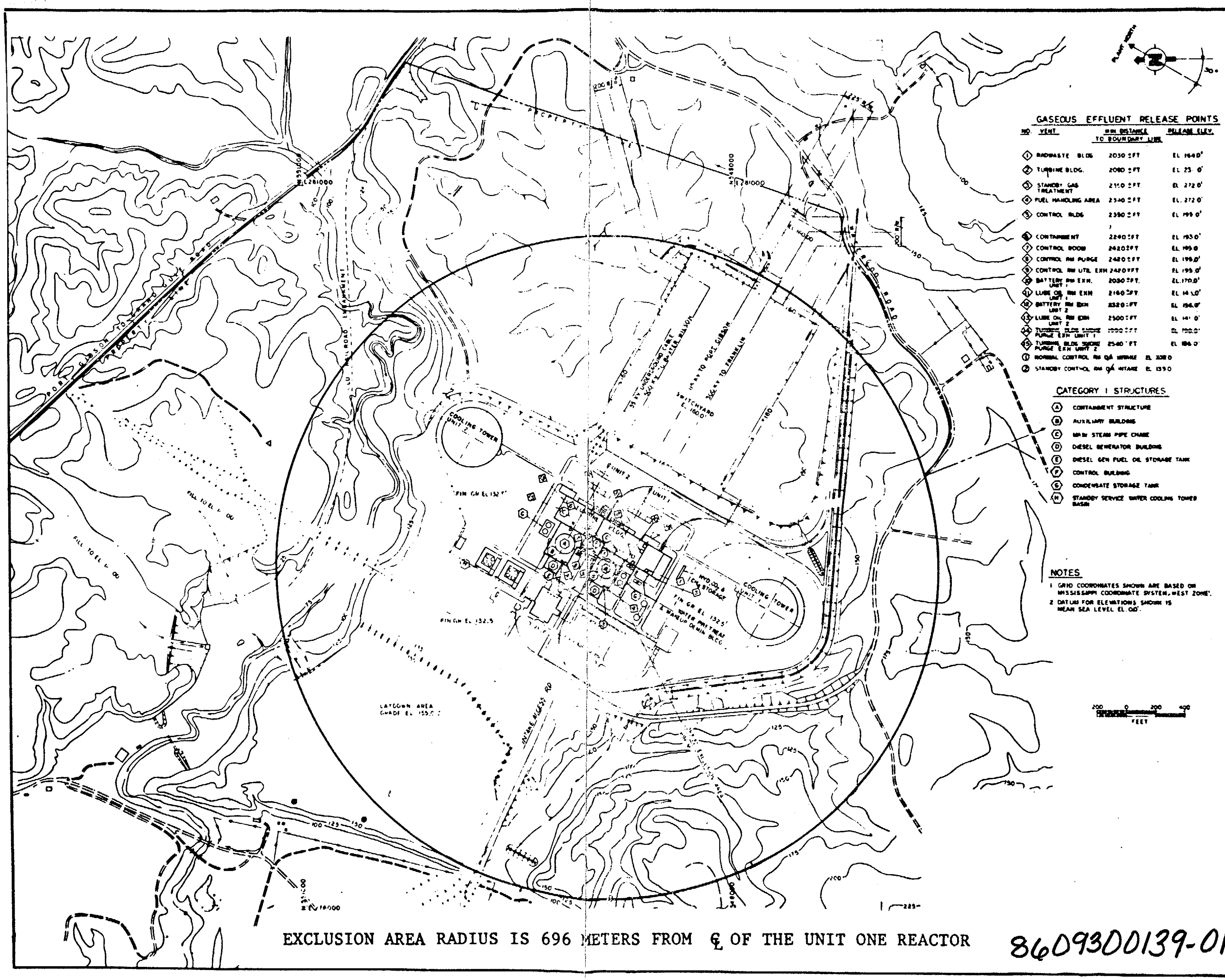
The Commission made proposed determinations that the amendment involves no significant hazards consideration which were published in the Federal Register (51 FR 18685) on May 21, 1986 and (51 FR 20370) on June 4, 1986, and consulted with the state of Mississippi did not have any comments.

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and the security nor the the health and safety of the public.

Principal Contributors:

M. McCoy, Reactor Systems Branch, DBL
H. Li, Electrical Instrumentation and Control Systems Branch, DBL
G. Suh, Reactor Systems Branch, DBL
J. Stang, BWR Project Directorate No. 1, DBL
L. Kintner, BWR Project Directorate No. 4, DBL

Dated: September 23, 1986



TI APERTURE CARD

Also Available On Aperture Card

FIGURE 5.1.1-1
EXCLUSION AREA AND GASEOUS EFFLUENT RELEASE POINTS