August 27, 1986

Docket No.: 50-416

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

Dear Mr. Kingsley:

SUBJECT: LICENSE AMENDMENT CHANGING LICENSE CONDITION 2.C.(20) AND TECHNICAL SPECIFICATIONS

RE: GRAND GULF NUCLEAR STATION, UNIT 1

The Commission has issued the enclosed Amendment No. 18 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. This amendment consists of changes to License Condition 2.C.(20) and to the Technical Specifications (TSs) in response to your application dated May 19, 1986 as supplemented by letters dated July 25 and July 29, 1986.

This amendment adds an under voltage protection device for the Division 3 diesel generator, changes License Condition 2.C.(20) and the TSs to allow modifications to be made to the standby service water system, clarifies the TSs regarding breaker response time for the end-of-life recirculation pump trip system, and changes the test loads for the batteries. Changes to License Condition 2.C.(20) and Technical Specification Pages 1-3, 3/4 3-42, 3/4 7-2 and B 3/4 3-3 are effective upon issuance of this amendment. Changes on Technical Specification Pages 3/4 3-29, 3/4 3-32a, 3/4 3-35, 3/4 8-6 and 3/4 8-12 are effective when the equipment necessitating the changes on those pages is installed and operable. For those Technical Specification changes that are not effective upon issuance, you are requested to inform the NRR by letter of their effective dates within 30 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. 18 to License No. NPF-29
2. Safety Evaluation

cc w/enclosures:
See next page

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PD#4/LA	PD#4/PM	PSB/C	FOB/D	OGC	PD#4/D
MO'Brien	LKintner:lb	JHulman	DVassallo	Young	WButler
8/20/86	8/20/86	8/21/86	Not Required /86	8/20 /86	8/27/86

Mr. Oliver D. Kingsley, Jr.
Mississippi Power & Light Company

Grand Gulf Nuclear Staiton

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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. 18
License No. NPF-29

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

   A. The application for amendment by Mississippi Power & Light Company, Middle South Energy, Inc., and South Mississippi Electric Power Association, (the licensees) dated May 19, 1986 as supplemented by letters dated July 25 and July 29, 1986, 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, Facility Operating License NPF-29 is amended as follows:

   A. Change paragraph 2.C.(20) to read as follows:

   (20) Standby Service Water System (Section 9.2.1 SER, SSER #2)

   No irradiated fuel may be stored in the Unit 1 spent fuel storage pool prior to completion of modifications to either loop A or loop B of the standby service water (SSW) system and verification that the design flow can be achieved to all essential SSW system components in the modified loop. However, should a core offloading be necessary prior to completion of these modifications (scheduled for the first refueling outage), irradiated fuel may be placed in the

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spent fuel pool when the RHR system operating in the spent fuel pool cooling mode is available. Until the SSW loops are modified, the spent fuel pool cooler in an unmodified loop shall be isolated from the loop by locked closed valves or the loop shall be declared inoperable. The position of these valves shall be verified every 31 days until the design flowrate for the SSW loop is demonstrated. The surveillance to be performed is to verify that any unmodified SSW loop with valves which are not locked closed is declared inoperable.

3. The license is further 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. 18 , 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.

4. License Condition 2.C.(20) and Technical Specification Pages 1-3, 3/4 3-42, 3/4 7-2, and B 3/4 3-3 in this amendment are effective upon issuance of this amendment. Technical Specification Pages 3/4 3-29, 3/4 3-32a, 3/4 3-35, 3/4 8-6 and 3/4 8-12 are effective when the equipment necessitating the changes on these Technical Specification pages is installed and made operable.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: August 27, 1986

PD#4/LA	PD#4/PM	OGC	PD#4/D
MO'Brien	LKintner:lb		WButler
6/10/86	8/18/86	/22/86	8/27/86

ATTACHMENT TO LICENSE AMENDMENT NO. 18

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
1-3	1-3
1-4	1-4*
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30*
--	3/4 3-32a
3/4 3-35	3/4 3-35
3/4 3-36	3/4 3-36*
3/4 3-41	3/4 3-41*
3/4 3-42	3/4 3-42
3/4 7-1	3/4 7-1*
3/4 7-2	3/4 7-2
3/4 8-5	3/4 8-5*
3/4 8-6	3/4 8-6
3/4 8-11	3/4 8-11*
3/4 8-12	3/4 8-12
B 3/4 3-3	B 3/4 3-3
B 3/4 3-3a	B 3/4 3-3a*

E-AVERAGE DISINTEGRATION ENERGY

1.11 E shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.12 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation 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 any series of sequential, overlapping or total steps such that the entire response time is measured.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.13 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker from initial movement of the associated:

a. Turbine stop valves, and
b. Turbine control valves.

The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured, except for the breaker arc suppression time which is not measured but is validated by surveillance tests to conform to the manufacturer's design value.

FRACTION OF LIMITING POWER DENSITY

1.14 The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the limiting LHGR for that bundle type.

FRACTION OF RATED THERMAL POWER

1.15 The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

FREQUENCY NOTATION

1.16 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM

1.17 The GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM is the system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

The OPERABILITY of the control rod block instrumentation in OPERATIONAL CONDITION 5 is to provide diversity of rod block protection to the one-rod-out interlock.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION[a]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
C. DIVISION 3 TRIP SYSTEM			
1. HPCS SYSTEM			
a. Reactor Vessel Water Level - Low, Low, Level 2	4[b]	1, 2, 3, 4*, 5*	33
b. Drywell Pressure - High##	4[b]	1, 2, 3	33
c. Reactor Vessel Water Level-High, Level 8	2[c]	1, 2, 3, 4*, 5*	31
d. Condensate Storage Tank Level-Low	2[d]	1, 2, 3, 4*, 5*	34
e. Suppression Pool Water Level-High	2[d]	1, 2, 3, 4*, 5*	34
f. Manual Initiation##	1	1, 2, 3, 4*, 5*	32
D. LOSS OF POWER			
1. Division 1 and 2			
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	4	1, 2, 3, 4**, 5**	30
b. 4.16 kV Bus Undervoltage (BOP Load Shed)	4	1, 2, 3, 4**, 5**	30
c. 4.16 kV Bus Undervoltage (Degraded Voltage)	4	1, 2, 3, 4**, 5**	30
2. Division 3			
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	4	1, 2, 3, 4**, 5**	30
b. 4.16 kV Bus Undervoltage (Degraded Voltage)	4	1, 2, 3, 4**, 5**	30

(a) A channel may be placed in an inoperable status for up to 2 hours during periods of required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(b) Also actuates the associated division diesel generator.

(c) Provides signal to close HPCS pump discharge valve only.

(d) Provides signal to HPCS pump suction valves only.

* Applicable when the system is required to be OPERABLE per Specification 3.5.2 or 3.5.3.

** Required when applicable ESF equipment is required to be OPERABLE.

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 135 psig.

\## Prior to STARTUP following the first refueling outage, the injection function of Drywell Pressure - High and Manual Initiation are not required to be OPERABLE with indicated reactor vessel water level on the wide range instrument greater than Level 8 setpoint coincident with the reactor pressure less than 600 psig.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months, during shutdown, by verifying that either:

1. The battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for 4 hours for Divisions 1 and 2 and 2 hours for Division 3 when the battery is subjected to a battery service test, or

2. The battery capacity is adequate to supply a dummy load of the following profile, which is verified to be greater than the actual emergency load, while maintaining the battery terminal voltage greater than or equal to 105 volts.

a) Division 1

$\geq$950 amperes for the first 60 seconds
$\geq$228 amperes for the next 119 minutes
$\geq$399 amperes for the next 60 seconds
$\geq$228 amperes for the next 118 minutes
$\geq$416 amperes for the last 60 seconds

b) Division 2

$\geq$462 amperes for the first 60 seconds
$\geq$221 amperes for the next 119 minutes
$\geq$392 amperes for the next 60 seconds
$\geq$221 amperes for the next 118 minutes
$\geq$278 amperes for the last 60 seconds

c) Division 3

$\geq$76 amperes for the first 60 seconds
$\geq$16 amperes for the next 59 minutes
$\geq$18 amperes for the last 60 minutes

e. At least once per 60 months during shutdown by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60 month interval, this performance discharge test may be performed in lieu of the battery service test.

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
D. LOSS OF POWER (Continued)		
2. Division 3 (Continued)		
b. 4.16 kV Bus Undervoltage (Degraded Voltage)	1. 4.16 kV Basis 3661 volts	3661 ± 102.5 volts
	2. 120 volt Basis 104.6 volts	104.6 ± 2.93 volts
	3. Time Delay	
	5 minutes/No LOCA	5 minutes ± 30 seconds
	4 seconds/LOCA	(4.0 ± 0.4 seconds)

SURVEILLANCE REQUIREMENTS (Continued)

9. Verifying the diesel generator operates for at least 24 hours. Diesel generators 11 and 12 shall be loaded to greater than or equal to 5450 kW but not to exceed 5740 kW for 24 hours. Diesel generator 13 shall be loaded to greater than or equal to 3630 kW for the first 2 hours of this test and to 3300 kW during the remaining 22 hours. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24-hour test, perform Surveillance Requirement 4.8.1.1.2.d.7.a).2) and b).2)*.

10. Verifying that the auto-connected loads to each diesel generator do not exceed 5740 kW for diesel generators 11 and 12 and 3300 kW for diesel generator 13.

11. Verifying the diesel generator's capability to:

    a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

    b) Transfer its loads to the offsite power source, and

    c) Be restored to its standby status.

12. Verifying that with the diesel generator operating in a test mode and connected to its bus that a simulated ECCS actuation signal overrides the test mode by returning the diesel generator to standby operation.

13. [DELETED]

14. [DELETED]

15. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval for diesel generators 11 and 12.

---

*If Surveillance Requirement 4.8.1.1.2.d.7.a)2) or b)2) are not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test. Instead, the diesel generator may be operated at the load specified by Surveillance Requirement 4.8.1.1.2.a.5 for one hour or until operating temperatures have stabilized.

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
B. DIVISION 2 TRIP SYSTEM (Continued)				
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R[a]	1, 2, 3
b. Drywell Pressure-High	S	M	R[a]	1, 2, 3
c. ADS Timer	NA	M	Q	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3	S	M	R[a]	1, 2, 3
e. LPCI Pump B and C Discharge Pressure-High	S	M	R[a]	1, 2, 3
f. Manual Initiation	NA	R[b]	NA	1, 2, 3
C. DIVISION 3 TRIP SYSTEM				
1. HPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low, Level 2	S	M	R[a]	1, 2, 3, 4*, 5*
b. Drywell Pressure-High##	S	M	R[a]	1, 2, 3
c. Reactor Vessel Water Level-High, Level 8	S	M	R[a]	1, 2, 3, 4*, 5*
d. Condensate Storage Tank Level - Low	S	M	R[a]	1, 2, 3, 4*, 5*
e. Suppression Pool Water Level - High	S	M	R[a]	1, 2, 3, 4*, 5*
f. Manual Initiation##	NA	R[b]	NA	1, 2, 3, 4*, 5*
D. LOSS OF POWER				
1. Division 1 and 2				
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	NA	M[e]	R	1, 2, 3, 4**, 5**
b. 4.16 kV Bus Undervoltage (BOP Load Shed)	NA	M[e]	R	1, 2, 3, 4**, 5**
c. 4.16 kV Bus Undervoltage (Degraded Voltage)	NA	M[e]	R	1, 2, 3, 4**, 5**
2. Division 3				
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	NA	NA	R	1, 2, 3, 4**, 5**
b. 4.16 kV Bus Undervoltage (Degraded Voltage)	NA	NA	R	1, 2, 3, 4**, 5**

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

d. In OPERATIONAL CONDITION 5 with the SSW subsystem, which is associated with an RHR system required OPERABLE by Specification 3.9.11.1 or 3.9.11.2, inoperable, declare the associated RHR system inoperable and take the ACTION required by Specification 3.9.11.1 or 3.9.11.2, as applicable.

e. In OPERATIONAL CONDITION *, with the SSW subsystem, which is associated with a diesel generator required OPERABLE by Specification 3.8.1.2, inoperable, declare the associated diesel generator inoperable and take the ACTION required by Specification 3.8.1.2. The provisions of Specification 3.0.3 are not applicable.

f. In OPERATIONAL CONDITIONS 1, 2, 3, 4, or 5 with the SSW subsystem, which is associated with a diesel generator required OPERABLE by Specification 3.8.1.1 or 3.8.1.2, inoperable, declare the associated diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2 as applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 At least the above required standby service water system subsystem(s) shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown by verifying that each automatic valve servicing safety related equipment actuates to its correct position on an actuation test signal.

INSTRUMENTATION

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 40% of RATED THERMAL POWER.

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within one hour.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:

   1. If the inoperable channels consist of one turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within one hour.

   2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or reduce THERMAL POWER to less than 40% of RATED THERMAL POWER within the next 6 hours.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or reduce THERMAL POWER to less than 40% of RATED THERMAL POWER within the next 6 hours.

4.3.4.2.1 Each end-of-cycle recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.2.1-1.

4.3.4.2.2. LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.4.2.3 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME of each trip function shown in Table 3.3.4.2-3 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include two turbine control valve channels from one trip system and two turbine stop valve channels from the other trip system such that all channels are tested at least once per 36 months. The time required for Breaker Interruption* shall be verified at least once per 60 months and added to the 18-month trip system times to verify that the overall END-OF-CYCLE RECIRCULATION PUMP TRIP RESPONSE TIME is within its limit.

*Breaker Interruption time is defined as Breaker Response time plus Arc Suppression time. Breaker Response is the time from application of voltage to the trip coil until the main contacts separate. Arc Suppression is the time from main contact separation until the complete suppression of the electrical arc across the open contacts. Breaker Response shall be verified by testing and added to the manufacturer's design Arc Suppression time of 12 ms to determine Breaker Interruption time. The breaker arc suppression time shall be validated by the performance of periodic contact gap measurements and high potential tests on the breaker vacuum interrupters in accordance with plant procedures.

3/4.7 PLANT SYSTEMS

3/4.7.1 SERVICE WATER SYSTEMS

STANDBY SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.1 Each of the following independent standby service water (SSW) system subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE SSW pump, and
b. An OPERABLE flow path capable of taking suction from the associated SSW cooling tower basin and transferring the water through the RHR heat exchangers and to associated plant equipment, as required, shall be OPERABLE as follows:

   1. In OPERATIONAL CONDITIONS 1, 2, and 3: two subsystems; and
   2. In OPERATIONAL CONDITIONS 4, 5, and *: the subsystems associated with the systems and components required to be OPERABLE by Specifications 3.4.9.2, 3.5.2, 3.8.1.2, 3.9.11.1 or 3.9.11.2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5 and *.

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3:

   1. With one SSW subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
   2. With both SSW subsystems inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN** within the following 24 hours.

b. In OPERATIONAL CONDITION 3 or 4 with the SSW subsystem, which is associated with an RHR loop required OPERABLE by Specification 3.4.9.1 or 3.4.9.2, inoperable, declare the associated RHR loop inoperable and take the ACTION required by Specification 3.4.9.1 or 3.4.9.2, as applicable.

c. In OPERATIONAL CONDITION 4 or 5 with the SSW subsystem, which is associated with an ECCS pump required OPERABLE by Specification 3.5.2, inoperable, declare the associated ECCS pump inoperable and take the ACTION required by Specification 3.5.2.

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*When handling irradiated fuel in the primary or secondary containment.

**Whenever both SSW subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

NOTATION

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 135 psig.

\## Prior to STARTUP following the first refueling outage, the injection function of Drywell Pressure - High and Manual Initiation are not required to be OPERABLE with indicated reactor vessel water level on the wide range instrument greater than Level 8 setpoint coincident with the reactor pressure less than 600 psig.

* Applicable when the system is required to be OPERABLE per Specification 3.5.2 or 3.5.3.

** Required when ESF equipment is required to be OPERABLE.

(a) Calibrate trip unit at least once per 31 days.

(b) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days as a part of circuitry required to be tested for automatic system actuation.

(c) DELETED

(d) DELETED

(e) Functional Testing of Time Delay Not Required

5. Verifying that on an ECCS actuation test signal, without loss of offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

6. [DELETED]

7. Simulating a loss of offsite power in conjunction with an ECCS actuation test signal, and:

   a) For Division 1 and 2:

      1) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

      2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ± 416 volts and 60 ± 1.2 Hz during this test.

   b) For Division 3:

      1) Verifying de-energization of the emergency bus.

      2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency bus with the permanently connected loads within 10 seconds and the autoconnected emergency loads within 20 seconds and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency bus shall be maintained at 4160 ± 416 volts and 60 ± 1.2 Hz during this test.

8. Verifying that all automatic diesel generator trips are automatically bypassed upon an ECCS actuation signal except:

   a) For Divisions 1 and 2, engine overspeed, generator differential current, and low lube oil pressure.

   b) For Division 3, engine overspeed and generator differential current.

4.8.2.1 Each of the above required 125-volt batteries and chargers shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

   1. The parameters in Table 4.8.2.1-1 meet the Category A limits, and

   2. Total battery terminal voltage is greater than or equal to 129-volts on float charge.

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110-volts, or battery overcharge with battery terminal voltage above 150-volts, by verifying that:

   1. The parameters in Table 4.8.2.1-1 meet the Category B limits,

   2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohms, and

   3. The average electrolyte temperature of every sixth connected cells is above 60°F.

c. At least once per 18 months by verifying that:

   1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

   2. The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material,

   3. The resistance of each cell and terminal connection is less than or equal to 150×10^{-6} ohms, and

   4. The battery charger will supply:

      a) For Divisions 1 and 2, at least 400 amperes at a minimum of 125 volts for at least 10 hours.

      b) For Division 3, at least 50 amperes at a minimum of 125 volts for at least 4 hours.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. With one channel inoperable, place the inoperable channel in the tripped condition within one hour* or declare the associated system(s) inoperable.

b. With more than one channel inoperable, declare the associated system(s) inoperable.

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, declare the associated ADS trip system or ECCS inoperable.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 8 hours or declare the associated ADS trip system or ECCS inoperable.

ACTION 33 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel(s) in the tripped condition within one hour* or declare the HPCS system inoperable.

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour* or declare the HPCS system inoperable.

*The provisions of Specification 3.0.4 are not applicable.

RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION (Continued)

feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a closure sensor for each of two turbine stop valves provides input to one EOC-RPT system; a closure sensor from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 40% of RATED THERMAL POWER are annunciated in the control room. The automatic bypass setpoint is feedwater temperature dependent due to the subcooling changes that affect the turbine first-stage pressure-reactor power relationship. For RATED THERMAL POWER operation with feedwater temperature greater than or equal to 420°F, an allowable setpoint of $\leq$ 26.9% of control valve wide open turbine first-stage pressure is provided for the bypass function. This setpoint is also applicable to operation at less than RATED THERMAL POWER with the correspondingly lower feedwater temperature. The allowable setpoint is reduced to $\leq$ 22.5% of control valve wide open turbine first-stage pressure for RATED THERMAL POWER operation with a feedwater temperature between 370°F and 420°F. Similarly, the reduced setpoint is applicable to operation at less than RATED THERMAL POWER with the correspondingly lower feedwater temperature.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 190 ms. Included in this time are: the response time of the sensor, the response time of the system logic and the breaker interruption time. Breaker interruption time includes both breaker response time and the manufacturer's design arc suppression time of 12 ms.

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.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

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.

IDENTIFIED LEAKAGE

1.18 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

1.19 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

LIMITING CONTROL ROD PATTERN

1.20 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.21 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.22 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e., all relays and contacts, all trip units, solid state logic elements, etc., of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.23 The MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be the highest value of the FLPD which exists in the core.

MEMBER(S) OF THE PUBLIC

1.24 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 18 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 May 19, 1986, as supplemented by letters dated July 25 and July 29, 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: (1) change Technical Specifications Tables 3.3.3-1, 3.3.3-2, and 4.3.3.1-1 to include an additional undervoltage protection device for the Division 3 emergency power bus and change Surveillance Requirement 4.8.1.1.2.d.12 to include an emergency override of the test mode for the Division 3 diesel generator; (2) change License Condition 2.C.(20) and Surveillance Requirement 4.7.1.1.a.2 to allow modification of the standby service water (SSW) loop A so that it will provide design water flows to all essential components; (3) clarify Surveillance Requirement 4.3.4.2.3, "End-of-Cycle Recirculation Pump Trip System Instrumentation," and associated bases and definition regarding the testing of breaker response time; and (4) change Surveillance Requirement 4.8.2.1.d.2 for the batteries and battery chargers by increasing test loads to allow increased loading on the batteries.

2.0 EVALUATION

2.1 Addition of an Undervoltage Protection Device and Revision of Emergency Override of the Test Mode for the Division 3 Diesel Generator

The proposed design changes associated with this Technical Specification change are made to fulfill License Condition 2.C.(37) parts a and b. One design change is the addition of a new first level (degraded voltage) of undervoltage protection which protects the high pressure core spray (HPCS) equipment from a sustained degraded voltage condition. The existing undervoltage protection devices will become the second level (loss of voltage) of protection. The first level protection is set at 88% of 4.16 kV to sense a degraded voltage. There are two specific elements, one associated with non-accident conditions and the other associated with accident conditions. The non-accident portion of the design provides a 5-minute time delay setting on the high pressure core spray (HPCS) bus when the bus voltage is at or below 88% of the nominal offsite power source voltage. An HPCS system undervoltage alarm in the main control room will be sounded and a 5-minute time delay will be initiated. After 5 minutes of sustained undervoltage, breaker Nos. 4, 5, and 6 will be tripped and the HPCS bus will be fed from the HPCS diesel generator. During these 5 minutes, if the bus voltage drops below 72%, breaker Nos. 4, 5, and 6 will be tripped by the level 2 protection device. Analyses performed for the HPCS pump motor and 480 volt motor-operated valves have shown that a voltage drop to 72% for 5 minutes is not likely to damage or reduce the life expectancy of the equipment. The only other equipment connected to the Division 3 bus that could be adversely affected by undervoltage is the battery charger. Below 85% of nominal voltage, the output and input current of the battery charger will decrease, but there is no thermal or other damage to the charger. The Division 3 battery will still handle the DC load for at least two hours. During a loss of coolant accident (LOCA), after a bus undervoltage of 88% for 4 seconds, the undervoltage protection device will initiate a HPCS system undervoltage alarm and also cause the auxiliary relay to bypass the 5-minute timing relay. The HPCS bus will be fed from the HPCS diesel generator.

The other design change is the addition of an emergency override provision in a test mode which makes the Division 3 design the same as the design for Divisions 1 and 2. With the diesel generator operating in a test mode and connected to its bus, if any of the three breakers are closed and a LOCA occurs, the diesel generator bus feeder breaker is tripped and the diesel generator remains running in standby operation. This design feature is in conformance with position 1(3) of Regulatory Guide 1.108. "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants."

The design changes associated with this proposed Technical Specification change provide assurance that the loads connected to the Division 3 bus will trip before possible damage from a sustained undervoltage condition occurs and that the Division 3 diesel generator returns to the standby mode when an emergency signal is detected. The staff concludes that the design changes associated with this proposed Technical Specification change are acceptable. The staff further concludes that the proposed change to Technical Specifications is acceptable.

2.2 Change License Condition 2.C.(20) and Surveillance Requirement 4.7.1.1.a.2 to Allow Modification of Standby Service Water Loop "A"

The present license condition requires that both "A" and "B" loops of the SSW system be modified and their flow rates verified before placing spent fuel into the spent fuel pool. As identified in the submittal dated May 19, 1986, the licensee modified the "B" loop of the SSW system during a fall 1985 outage. During this fall outage, all spent fuel was contained within the reactor vessel. The licensee intends to take the "A" loop out of service during the first refueling outage. In order to meet the license condition during this outage, the licensee would have to modify the "A" loop of the SSW system prior to moving spent fuel. This would result in a prolonged refueling outage. In order to minimize the outage time, the licensee has requested in a May 19, 1986, submittal to change the license condition so that both SSW loops would not be required to be modified and verified prior to placing spent fuel into the spent fuel pool. Instead, the requirement would apply only to the "A" loop or the "B" loop during refueling.

Both loops are to be operational upon startup. The proposed license condition does not state explicitly that it is applicable only until the end of the first refueling outage. In order to assure that the license condition is used only for this interim period, the licensee committed in a submittal dated July 25, 1986, to request deletion of proposed License Condition 2.C.(20) within 90 days following restart after the first refueling.

The licensee has performed an analysis in support of the adequacy of a single loop of the SSW system. This analysis is intended to demonstrate that the cooling capacity and water inventory of the SSW loop and cooling tower basin B alone is capable of dissipating the heat from accidents during refueling outage one. In order to bound the heat load that would have to be dissipated from the reactor and spent fuel pool, the licensee made the following assumptions:

(1) Unit 1 experiences a loss of coolant accident at full power coincident with a loss of offsite power;

(2) Unit 2 is not operational, and

(3) The spent fuel pool contains no fuel from previous refuelings.

The staff agrees with the licensee that the heat associated with an accident at full power, including the decay heat associated with a full reactor core, bounds the heat load that would have to be dissipated during the first refueling outage. This is true even with spent fuel in the spent fuel pool because the new fuel inside the reactor will not have been irradiated and will not be shedding decay heat, and the heat load from irradiated fuel in the vessel will be relatively low.

In addition to the conservative heat load assumptions, the licensee also assumed that:

(1) SSW loop A is removed from service and SSW basin A is drained (inventory transfer between basins is not possible);

(2) makeup to the SSW system is not available from normal means of supply; and

(3) The accident occurred during the worst 30-day meteorology in regard to water consumption.

These assumptions are in conformance with the guidelines of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants."

The analysis for the first refueling outage resulted in a maximum water loss estimate of 6.21 x 106 gallons over 30 days. Since each basin contains about 6.6 x 106 gallons, the analysis demonstrates that the water inventory in basin B alone is enough to dissipate safely the heat associated with any accident that could occur during the first refueling outage.

Although only two tower cells are assumed to be operable, the temperature range (i.e., the cooling tower water temperature increase due to the plant heat load) during the maximum heat rejection period is still less than that estimated in the two unit analysis in the FSAR. Hence, a worst day analysis for maximum temperature, as suggested in Regulatory Guide 1.27, was not required.

The licensee also performed an analysis of a LOCA in Unit 1 with fuel from 18 refuelings in the spent fuel pool. For this analysis it was assumed that the loop A SSW pump and piping had been replaced and water transfer between the basins was possible. This analysis was intended to bound all accident conditions associated with Unit 2 not yet in operation, and after the first refueling outage for Unit 1. What makes this period important is that with Unit 2 not yet in service, only two cells in each tower are operable, and with an assumed diesel failure, the heat dissipation capacity is still the same as it was for the first refueling outage analysis. However, the water inventory is greater. For the same critical 30-day period as was assumed for the first refueling outage period, the total evaporation and drift losses were determined to total 7.8 x 106 gallons, as compared to the available inventory of 10.3 x 106 gallons in the two basins.

When the staff reviewed this calculation, it was noted that the cooling tower temperature range for this case during the period of maximum heat rejection, appeared to be greater than that estimated in the FSAR analysis for Units 1 and 2. For this reason, the staff requested the licensee to calculate return flow temperatures during the worst one-day meteorology with respect to heat dissipation. The licensee complied, and estimated a maximum return temperature of 90.5°F. This estimate exceeds the design basis return temperature of 90°F. The licensee explained that this should not be a reason for concern on the basis that mixing with cooler water in the basin was ignored. Hence, the pump suction temperature was assumed to be simultaneously the same as the basin return temperature. Actually, there will be a significant time lag due to mixing and the thermal inertia of the large water inventory relative to the flow. If mixing of the water from the cooling towers with the cooler basin water were considered, the calculated return water temperature would be less than 90°F. Therefore, the staff concludes that the design basis return water temperature of 90°F should not be reached.

Technical Specification 4.7.1.1.a.2 requires the verification that valves which isolate the spent fuel pool cooling system from the SSW system are locked closed. During the refueling outage, the B loop spent fuel pool cooler will be in service and the A loop spent fuel pool cooler will be undergoing flow testing. After the modifications are completed, there is no need for the valves which isolate the spent fuel pool cooling system from the SSW system. The licensee has proposed the deletion of this requirement to be effective at the start of the refueling outage. This is acceptable to the staff because one of the loops has been modified and verified to deliver the design flow rates to the safety-related components and the other loop (i.e., the A loop, via the spent fuel pool cooling system isolation valves) must be opened after modifications for testing. At startup from the first refueling outage, this surveillance requirement should not be applicable for either loop of the SSW because both will have been modified and tested to deliver design flow rates.

In summary, the staff's review of the licensee's analysis indicates a relatively low heat load anticipated during the first refueling outage, and an adequate heat dissipation capability available through the B loop. Therefore, the staff concludes that the draining of SSW system basin A during the first refueling outage will not result in a loss of adequate cooling capability in accordance with the guidelines of Regulatory Guide 1.27. In light of the licensee's commitment to request elimination of the proposed License Condition 2.C.(20) within 90 days of startup following the first refueling, we conclude that the proposed modification to License Condition 2.C.(20) is acceptable. Since Technical Specification 4.7.1.1.a.2 is not required following the modification of SSW loop A, the staff concludes that its deletion thereafter is acceptable.

2.3 Clarification of End-of-Cycle Recirculation Pump Trip System Breaker Response Time Measurement

The proposed changes provide clarification of the terminology and intent of Surveillance Requirement 4.3.4.2.3 for the end-of-cycle recirculation pump trip (EOC-RPT) system response time. As currently written, the Technical Specification requires verification of an arc suppression time of 50 millisecond (msec). The manufacturer's typical design value for breaker arc suppression time is 12 msec. While arc suppression time cannot be physically measured in situ, the manufacturer's design value can be validated by performance of contact gap measurements and high potential tests on the breaker vacuum interrupters. The revised Surveillance Requirement would require this validation.

The terminology used in the present Technical Specifications, "breaker arc suppression time," was intended to be the time interval from application of voltage to the trip coil until the complete suppression of the electrical arc across the open breaker contacts. The licensee stated that the present requirement to determine breaker arc suppression time is impractical to perform due to the nature of the test. The test would require two high speed trips for each reactor recirculation pump at pump speeds above 40%, thus subjecting the plant to additional undesired transients. Alternate methods using dummy loads to test the breakers would require extensive plant modifications. A timing test for arc suppression is not recommended by vendor manual GEK-396711-D. Since vacuum interrupter integrity and contact gap measurements are the primary factors for establishing the design arc suppression time of 12 msec, a timing test for arc suppression is not necessary. Technical Specification 4.8.4.1.b requires that each primary containment penetration conductor overcurrent device shall be demonstrated operable at least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations. GGNS electrical procedure 06-EL-1B33-0-0001 requires a high potential test on vacuum interrupter and contact gap measurement for EOC-RPT breakers. The proposed change to Technical Specification 4.3.4.2.3 would require the response time of the breaker to be tested during the 60-month maintenance surveillance interval with the plant in a cold shutdown condition. Changes to the Bases on page B 3/4 3-3 reflect the changes made to Technical Specification 4.3.4.2.3. These changes do not have a significant effect on the EOC-RPT overall response time requirements (190 msec) of Technical Specification 4.3.4.2.3 and Table 3.3.4.2-3, as the typical breaker arc suppression item is only a small percentage of this time, i.e., 12 msec.

The staff concludes that the changes in definition (Page 1-3), Technical Specification 4.3.4.2.3 (Page 3/4 3-42) and Bases (Page B 3/4 3-3) are acceptable.

2.4 Increase of the Test Loads for Division 1 and Division 2 Batteries

Present Technical Specification 4.8.2.1.d.2 specifies an 18-month testing requirement to ensure that battery capacity is greater than the actual emergency load while maintaining the battery terminal voltage greater than or equal to 105 volts. At the first refueling outage, the loads on batteries A and B will be increased to allow for planned design modifications. These new loads will be added to the uninterruptible power supplies (UPS). These planned design modifications include:

(1) Changing the electrical supply for reactor protection system logic circuits and radiation and neutron monitoring circuits from the reactor protection bus to Class 1E UPS.

(2) Changing drywell atmosphere temperature monitoring circuits from the Class 1E ESF bus to the Class 1E UPS.

(3) Providing annunciation for the UPS inverters in the control room.

(4) Changing remote shutdown panel reactor core isolation cooling (RCIC) flow control loop from the ESF bus to the Class 1E UPS.

The proposed change to the Technical Specification 4.8.2.1.d.2 will allow load increases from 50 amps to 150 amps on Division 1 (battery A) and from 115 amps to 150 amps on Division 2 (battery B). The licensee stated that the load increase on Divisions 1 and 2 batteries has been evaluated to ensure that battery capacity has not been exceeded considering battery aging and temperature correction factors. Applicable provisions of IEEE Standards 485-1978 and 450-1980 were considered in developing these proposed changes.

The staff has reviewed the licensee's request to allow load increases on Division 1 and Division 2 batteries. The adequacy of the battery capacity will be assured by the periodic surveillance requirements on electrolyte level, float voltage, and specific gravity, in addition to the dummy load test. Since the acceptance criteria for these periodic surveillance tests have not been changed, the staff concludes that the proposed load increase on Divisions 1 and 2 batteries will not adversely impact the capability of Class 1E dc power system to perform its safety functions. Therefore, the staff finds the proposed changes to Technical Specification 4.8.2.1.d.2 (Page 3/4 8-12) acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes to requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. 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 a proposed finding 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 a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (51 FR 22239) on June 18, 1986, and consulted with the state of Mississippi. No public comments were received, and 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 to the health and safety of the public.

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