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

February 21, 1984

Docket No.: 50-416

Mr. J. P. McGaughy, Jr.
Vice President - Nuclear Production
Mississippi Power & Light Company
P. O. Box 1640
Jackson, Mississippi 39205

Dear Mr. McGaughy:

Subject: Amendment No. 12 to Facility Operating License No. NPF-13 - Grand Gulf Nuclear Station, Unit 1

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 12 to Facility Operating License No. NPF-13 for the Grand Gulf Nuclear Station, Unit 1. This amendment is in response to MP&L letters dated April 25, 1983, June 23, 1983, July 11, 1983, and September 12, 1983, which you submitted in partial response to the NRC Confirmation of Action (COA) letter of October 20, 1982. The COA letter called for MP&L to prepare and submit license amendment requests, where necessary, to correct administrative and technical deficiencies in your Technical Specifications during MP&L's review of the Grand Gulf Unit 1 surveillance procedures. This amendment grants changes to the Technical Specifications.

The changes to the Technical Specifications relate to Specifications 3.4.2.1, Low-Low Set Function; Tables 3.3.2-1 and 4.3.2.1-1, Low Condenser Vacuum; 4.3.4.2.3, EOC Breaker Arc Suppression Time; 4.3.7.6 and Table 3.3.6-2, Source Range Monitors; 4.7.6.1.3, Fire Pump Diesel Battery and 4.8.2.1, Load Profile for Division 2 Batteries. None of the changes involve a significant relaxation of the criteria used to establish safety limits or the bases for limiting safety system settings or limiting conditions for operation.

A copy of the related staff evaluation supporting Amendment No. 12 to Facility Operating License NPF-13 is enclosed. The Notice of Issuance will be included in the Commission's next regular monthly Federal Register Notice.

Sincerely,

Original signed by:

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclosures:
1. Amendment No. 12 to NPF-13
2. Staff Evaluation

cc w/ enclosures:
See next page

*SEE PREVIOUS CONCURRENCES

DL:LB#2/PM	DL:LB#2/LA	DL:LB#2/BC	OELD	~~DL:AD/L~~
DHouston*:pt	EGHylton*	ASchwencer	MWagner*	TMNovak
2/07/84	2/07/84	~~2/08/84~~ 2/21/84	2/~~17~~ 21/84	2/21/84

Grand Gulf

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Port Gibson, Mississippi 39150

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Jackson, Mississippi 39201

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Atlanta, Georgia 30309

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Jackson, Mississippi 39205

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

MISSISSIPPI POWER AND 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

License No. NPF-13
Amendment No. 12

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

   A. The applications for the amendment filed by the Mississippi Power and Light Company dated April 25, 1983, June 23, 1983, July 11, 1983, and September 12, 1983, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, the provisions of the Act, and the 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 as follows:

   A. Page changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) to read as follows:

      (2) The Technical Specifications contained in Appendix A, as revised through Amendment No. 12, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Attachments:
1. Changes to the Technical Specifications

Date of Issuance: February 21 , 1984

*SEE PREVIOUS CONCURRENCES

DL:LB#2/PM	DL:LB#2/LA	DL:LB#2/BC	OELD	DL:AD/L
MDHouston*:pt	EGHylton*	ASchwencer*	MWagner*	TMNovak
2/08/84	2/08/84	2/~~08~~ 21/84	2/21/84	2/21/84

ATTACHMENT TO LICENSE AMENDMENT NO. 12
FACILITY OPERATING LICENSE NO. NPF-13
DOCKET NO. 50-416

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. These revised pages are identified by Amendment number and contain a vertical line indicating the area of change.

REMOVE	INSERT
3/4 3-10	3/4 3-10
3/4/3-14	3/4 3-14
3/4 3-23a	3/4 3-23a
3/4 3-39	3/4-3-39
3/4 3-52	3/4 3-52
3/4 3-73	3/4 3-73
3/4 4-5	3/4 4-5
3/4 7-30	3/4 7-30
3/4 8-12	3/4 8-12

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level- Low Low, Level 2	6A, 7, 8, 10(c)(d)	2	1, 2, 3 and #	20
b. Reactor Vessel Water Level- Low Low Level 2 (ECCS - Division 3)	6B	4	1, 2, 3 and #	29
c. Reactor Vessel Water Level- Low Low Low, Level 1 (ECCS - Division 1 and Division 2)	5(n)	2	1, 2, 3 and #	29
d. Drywell Pressure - High	6A, 7(c)(d)	2	1, 2, 3	20
e. Drywell Pressure-High (ECCS - Division 1 and Division 2)	5(n)	2	1, 2, 3	29
f. Drywell Pressure-High (ECCS - Division 3)	6B	4	1, 2, 3	29
g. Containment and Drywell Ventilation Exhaust Radiation - High High	7	2(e)	1, 2, 3 and *	21
h. Manual Initiation	6A, 7, 8, 10(c)(d)	2	1, 2, 3 and *#	22
2. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level- Low Low Low, Level 1	1	2	1, 2, 3	20
b. Main Steam Line Radiation - High	1, 10(f)	2	1, 2, 3	23
c. Main Steam Line Pressure - Low	1	2	1	24
d. Main Steam Line Flow - High	1	2(g)	1, 2, 3	23
e. Condenser Vacuum - Low	1	2	1, 2,** 3**	23

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

ACTION

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 21 - Close the affected system isolation valve(s) within one hour or:
a. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
b. In Operational Condition *, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary containment and operations with a potential for draining the reactor vessel.

ACTION 22 - Restore the manual initiation function to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 23 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 24 - Be in at least STARTUP within 6 hours.

ACTION 25 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour.

ACTION 26 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable.

ACTION 27 - Close the affected system isolation valves within one hour and declare the affected system inoperable.

ACTION 28 - Lock the affected system isolation valves closed within one hour and declare the affected system inoperable.

ACTION 29 - Close the affected system isolation valves within one hour and declare the affected system or component inoperable or:
a. In OPERATIONAL CONDITION 1, 2 or 3 be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
b. In OPERATIONAL CONDITION # suspend CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

NOTES

* When handling irradiated fuel in the primary or secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

** The low condenser vacuum MSIV closure may be manually bypassed during reactor SHUTDOWN or for reactor STARTUP when condenser vacuum is below the trip setpoint to allow opening of the MSIVs. The manual bypass shall be removed when condenser vacuum exceeds the trip setpoint.

During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

(a) See Specification 3.6.4, Table 3.6.4-1 for valves in each valve group.

(b) A channel may be placed in an inoperable status for up to 2 hours for 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.

(c) Also actuates the standby gas treatment system.

(d) Also actuates the control room emergency filtration system in the isolation mode of operation.

(e) Two upscale-Hi Hi, one upscale-Hi Hi and one downscale, or two downscale signals from the same trip system actuate the trip system and initiate isolation of the associated containment and drywell isolation valves.

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
6. RHR SYSTEM ISOLATION (Continued)				
e. Drywell Pressure - High	S	M	R	1, 2, 3
f. Manual Initiation	NA	M[(a)]	NA	1, 2, 3

*When handling irradiated fuel in the primary or secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

**The low condenser vacuum MSIV closure may be manually bypassed during reactor SHUTDOWN or for reactor STARTUP when condenser vacuum is below the trip setpoint to allow opening of the MSIVs. The manual bypass shall be removed when condenser vacuum exceeds the trip setpoint.

#During CORE ALTERATION and operations with a potential for draining the reactor vessel.

(a) 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 part of circuitry required to be tested for automatic system isolation.

(b) Each train or logic channel shall be tested at least every other 31 days.

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

SURVEILLANCE REQUIREMENTS

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 at least the logic of one type of channel input, turbine control valve fast closure or turbine stop valve closure, such that both types of channel inputs are tested at least once per 36 months. The time allotted for breaker arc suppression, 50 ms, shall be verified at least once per 60 months.*

*Prior to STARTUP after the first refueling outage, the breaker arc suppression time of 12 ms, as determined by the manufacturer, shall apply.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD PATTERN CONTROL SYSTEM		
a. Low Power Setpoint	20 + 15, -0% of RATED THERMAL POWER	20 + 15, -0% of RATED THERMAL POWER
b. Intermediate Rod Withdrawal Limiter Setpoint	$\leq$ 70% of RATED THERMAL POWER	$\leq$ 70% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux-Upscale	$\leq$ 0.66 W + 42%*	$\leq$ 0.66 W + 45%*
b. Inoperative	NA	NA
c. Downscale	$\geq$ 5% of RATED THERMAL POWER	$\geq$ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	$\leq$ 12% of RATED THERMAL POWER	$\leq$ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq 1 \times 10^5$ cps	$\leq 1.5 \times 10^5$ cps
c. Inoperative	NA	NA
d. Downscale	$\geq$ 0.7 cps	$\geq$ 0.5 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq$ 108/125 of full scale	$\leq$ 110/125 of full scale
c. Inoperative	NA	NA
d. Downscale	$\geq$ 5/125 of full scale	$\geq$ 3/125 of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	$\leq$ 32 inches	$\leq$ 33.5 inches
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	$\leq$ 108% of rated flow	$\leq$ 111% of rated flow

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

INSTRUMENTATION

SOURCE RANGE MONITORS

LIMITING CONDITION FOR OPERATION

3.3.7.6 At least three source range monitor channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 2*, 3 and 4.

ACTION:

a. In OPERATIONAL CONDITION 2* with one of the above required source range monitor channels inoperable, restore at least 3 source range monitor channels to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4 with two or more of the above required source range monitor channels inoperable, verify all insertable control rods to be fully inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

SURVEILLANCE REQUIREMENTS

4.3.7.6 Each of the above required source range monitor channels shall be demonstrated OPERABLE by:

a. Performance of a:

   1. CHANNEL CHECK at least once per:

      a) 12 hours in CONDITION 2*, and

      b) 24 hours in CONDITION 3 or 4.

   2. CHANNEL CALIBRATION** at least once per 18 months.

b. Performance of a CHANNEL FUNCTIONAL TEST:

   1. Within 24 hours prior to moving the reactor mode switch from the Shutdown position, if not performed within the previous 7 days, and

   2. At least once per 31 days.

c. Verifying, prior to withdrawal of control rods, that the SRM count rate is at least 0.7 cps with the detector fully inserted.

*With IRM's on range 2 or below.
**Neutron detectors may be excluded from CHANNEL CALIBRATION.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 Of the following safety/relief valves, the safety valve function of at least 7 valves and the relief valve function of at least 6 valves other than those satisfying the safety valve function requirement shall be OPERABLE with the specified lift settings:

Number of Valves	Function	Setpoint* (psig)
8	Safety	1165 $\pm$ 11.6 psi
6	Safety	1180 $\pm$ 11.8 psi
6	Safety	1190 $\pm$ 11.9 psi
1#	Relief	1103 $\pm$ 15 psi
10	Relief	1113 $\pm$ 15 psi
9	Relief	1123 $\pm$ 15 psi

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the safety and/or relief valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 105°F, close the stuck open relief valve(s); if unable to close the open valve(s) within 2 minutes or if suppression pool average water temperature is 105°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more safety/relief tail-pipe pressure switches inoperable, restore the inoperable switch(es) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1.1 The tail-pipe pressure switch for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be 30 ± 5 psig by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and a
b. CHANNEL CALIBRATION at least once per 18 months.**

4.4.2.1.2 The relief valve function pressure actuation instrumentation shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST, including calibration of the trip unit, at least once per 31 days.
b. CHANNEL CALIBRATION, LOGIC SYSTEM FUNCTIONAL TEST and simulated automatic operation of the entire system at least once per 18 months.

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

#Initial opening of 1B21-F051B is 1103 $\pm$ 15 psig due to low-low set function.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-75, is within the acceptable limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water and sediment.

c. At least once per 18 months, during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.7.6.1.3 The diesel driven fire pump starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

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

1. The electrolyte level of each cell in each battery is above the plates, and

2. The overall battery set voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity for each cell is appropriate for continued service of the battery. The specific gravity, corrected to 77°F and full electrolyte level, shall be is greater than or equal to 1.20.

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

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

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

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$128 amperes for the next 119 minutes
$\geq$306 amperes for the next 60 seconds
$\geq$128 amperes for the next 118 minutes
$\geq$416 amperes for the last 60 seconds

b) Division 2

$\geq$427 amperes for the first 60 seconds
$\geq$186 amperes for the next 119 minutes
$\geq$357 amperes for the next 60 seconds
$\geq$186 amperes for the next 118 minutes
$\geq$243 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.

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

SAFETY EVALUATION
AMENDMENT NO. 12 TO NPF-13
GRAND GULF NUCLEAR STATION, UNIT 1
DOCKET NO. 50-416

Introduction

The licensee proposed changes to the Technical Specifications for Grand Gulf Unit 1 which are as follows:

(a) 3.4.2.1: Revises low-low set function for relief valves (July 11, 1983),

(b) Tables 3.3.2-1 and 4.3.2.1-1: Deletes automatic removal of low condenser vacuum bypass (June 23, 1983),

(c) 4.3.4.2.3: Postpones the surveillance test for breaker arc suppression time until first refueling outage (June 23, 1983),

(d) 4.3.7.6 and Table 3.3.6-2: Lowers the minimum allowable countrate for Source Range Monitor (SRM) Operability (September 12, 1983),

(e) 4.7.6.1.3: Revises surveillance procedure for fire pump diesel batteries (April 25, 1983), and

(f) 4.8.2.1: Increases the load profile for Division 2 125 volt DC batteries (June 23, 1983).

Evaluation

(a) 3.4.2.1: Low-Low Set Function for Relief Valves

The low-low setpoint relief system is designed to protect the containment from excessive loads by assuring that no more than one relief valve reopens subsequent to the first full blowdown on an isolation event. The system accomplishes this by releasing sufficient energy on the first relief so that a single valve has the capacity to accommodate subsequent necessary relieving of energy to the suppression pool. The staff review of the low-low function has been presented in Section 7.8(F) of the Safety Evaluation Report (SER) and Supplement No. 1 of the Safety Evaluation Report (SSER #1) issued in September 1981 and December 1981, respectively.

In the original design, this low-low logic was armed at a setpoint of 1113 psig. However, NSSS vendor analyses of loss of feedwater events indicated that an isolation transient may result in only 1 SRV (with 1103 psig setpoint) opening initially. The resultant pressure transient, with 1 SRV opened may not reach 1113 psig. Since the low-low set relief logic would not be armed in this scenario, a subsequent failure of the 1103 psig setpoint SRV later in the isolation event could result in the simultaneous opening of the 10 ganged SRVs with a setpoint of 1113 psig. To prevent the occurrence of this event, which was not included in the containment design analysis, a design modification to the subject logic was accomplished.

We have reviewed the functioning of the low-low setpoint relief system, as modified and find it acceptable. Therefore, we approve the changes to Section 3.4.2.1 of the Technical Specifications.

(b) Tables 3.3.2-1 and 4.3.2.1-1: Low Condenser Vacuum Bypass

In the current Technical Specifications, the present "**" note on Table 4.3.2.1-1 is in conflict with Specification 3.4.6.2 concerning maximum reactor steam dome pressure in OPERATIONAL CONDITION 2. The change will remove the exceptions (only when reactor steam dome pressure is greater than 1045 psig and/or turbine stop valves are open in OPERATIONAL CONDITIONS 2 and 3) to performing the Surveillance Requirements on the Condenser Vacuum - Low Trip function in OPERATIONAL CONDITIONS 2 and 3. The changes will also add the revised footnote to the Condenser Vacuum - Low Trip function in Table 3.3.2-1. OPERATIONAL CONDITIONS 2 and 3 refer to STARTUP and HOT STANDBY, respectively.

The "loss of condenser vacuum" transient is analyzed in the Final Safety Analysis Report (FSAR) Section 15.2.5. The transient analysis assumes that the plant is initially operating at 105 percent of nuclear boiler rated steam flow conditions (OPERATIONAL CONDITION 1 - POWER OPERATION). Condenser vacuum is assumed to be lost at a rate of 2 inches Hg per second. The turbine bypass valve and MSIV closure would follow main turbine and feedwater turbine trips about 5 seconds after they initiate the transient. FSAR Section 15.2.5.3.3 states that the effect of MSIV closure tends to be minimal since the closure of main turbine stop valves and subsequently the bypass valves have already shutoff the main steam line flow. This analysis demonstrates that even if the MSIV's do not close automatically from a low condenser vacuum signal (i.e., keylock switches inadvertently left in BYPASS), main steam line flow is shut off by closure of the turbine stop valve and bypass valves).

Since the transient analyzed in FSAR Section 15.2.5 considers OPERATIONAL CONDITION 1 as the worst case condition, bypassing the MSIV closure trip function in OPERATIONAL CONDITIONS 2 and 3 will also have a minimal effect on the transient. On the basis of a minimal effect and to achieve consistency within the Technical Specifications, we find these changes acceptable.

(c) 4.3.4.2.3: Breaker Arc Suppression Time

The changes for the breaker arc suppression time would recognize the test time determined by the manufacturer until a suitable test procedure and appropriate Technical Specifications are established prior to the first refueling outage. As currently written, the Technical Specifications would require verification of an arc suppression time of 50 ms while the manufacturer has determined by test that the actual arc suppression time for the Grand Gulf breaker is 12 ms. Rapid degradation of this time function is not expected, hence, the test frequency of once per 60 months. The current Technical Specifications incorrectly incorporate other parameters into the specified time value. The required total system response time of $\leq$ 190 ms in Table 3.3.4.2-3 remains unchanged. Thus, there is no effect on the accident analysis involving this system. Since the system response remains the same, we find the exception stated in the footnote to be acceptable until the first refueling outage.

(d) 4.3.7.6 and Table 3.3.6-2: Source Range Monitor

The changes for the Source Range Monitor (SRM) would lower the setpoint for down-scale trip to 0.7 count per second with a minimum allowable value of 0.5 per second and would define the OPERABILITY of the SRMs in terms of these lowered values. The 0.7 counts per second would be coupled with a procedural requirement for a signal to noise ratio of two or greater.

The SRM circuitry is used to provide information on the sub-critical multiplication of the core during startups. It is important to "see" the neutron population in the core, i.e., to be counting neutrons rather than noise, but the count rate is less important. The requirement for a signal to noise ratio of two or greater assures that neutrons are being counted. The value of 0.7 counts per second meets the guidance in Regulatory Guide 1.68 (Rev. 2) with respect to minimum count rates of startup.

The effect of a reduced neutron population ("core power") on the rod drop accident has been addressed by General Electric for the Cooper Station (letter J. M. Pilant to V. Moore dated April 4, 1974). This result has been confirmed by GE to also be applicable to Grand Gulf.

We have previously approved such changes in Technical Specifications for other BWR plants and thus, find this change to be acceptable for Grand Gulf.

(e) 4.7.6.1.3: Fire Pump Diesel Batteries

The changes for the Fire Pump Diesel Batteries are made to achieve consistency with Standard Technical Specifications and to delete the inappropriate reference to the pilot cell concept. Instead of pilot cell surveillances, suitable surveillances at 7 day and 92 day intervals are specified for each cell and the overall battery set. These changes do not diminish the level of confidence or reliability for this system. Since an equivalent component surveillance is provided, we find these changes acceptable.

(f) 4.8.2.1: Load Profile for Division 2 Batteries

The changes for the load profile for the Division 2 batteries are made to reflect the as-built condition of the plant and to prepare for a modification of the Division 2 inverter to a new class IE inverter. The Technical Specification surveillance requirements for the 125 volt DC Division 2 battery load profile are revised to accurately reflect the anticipated post LOCA load profile based upon the installed system loads. The actual connected loads for Division 2 will be greater than the present Technical Specification load profile. As a result, this change is necessary for Division 2. The change is based on the load profile for 100 percent utilization of the Division 2 Westinghouse inverters (i.e., the Division 2 Technical Specification load profile will be 77 amps higher than the actual connected loads). Since this testing profile is more stringent than that specified in the current Technical Specifications, we find the changes acceptable.

Environmental Consideration

The Commission has determined that the issuance of this amendment will not result in any environmental impacts other than those evaluated in the Final Environmental Statement since the activity authorized by the amendment is encompassed by the overall action evaluated in the Final Environmental Statement dated September 1981.

Conclusion

We have 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 security or to the health and safety of the public.

Dated: February 21 , 1984

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2/7/84	2/8/84

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