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

September 23, 1983

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. 10 to Facility Operating License No. NPF-13 - Grand Gulf Nuclear Station, Unit 1

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 10 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 June 14, 1983, June 23, 1983, and August 1, 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 and one time exceptions to some Technical Specifications for relief needed to restart the plant.

The changes to the Technical Specifications relate to Specifications Tables 3.3.3-1 and 4.3.3.1-1, Bases Figure 3/4 3-1, High Pressure Core Spray Operability and Table 3.6.4-1, RHR Jockey Pumps. The one time exceptions to the Technical Specifications relate to Specifications 4.4.2.1, 4.4.2.2 and Table 3.3.3-1, ADS Trip System, and 4.1.3.1, Scram Discharge Volume. 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. 10 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,

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

Enclosures:
1. Amendment No. 10
2. Staff Evaluation

cc w/ enclosures:
See next page

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* See Previous Concurrences

DL:LB#2/PM*	DL:LB#2/LA*	DL:LB#2/BG*	OELD*	DL:AD/L*
MDHouston:pt	EGHylton	ASchwencer		TMNovak
9/19 /83	9/19/83	9/13 /83	9/20 /83	9/19/83

A copy of the related staff evaluation supporting Amendment No. 10 to Facility Operating License NPF-13 is enclosed. Also enclosed is a copy of a related notice which has been forwarded to the Office of the Federal Register for publication.

Sincerely,

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

Enclosures:
1. Amendment No. 10 to NPF-13
2. Staff Evaluation
3. Federal Register Notice

cc w/ enclosures:
See next page

DL:LB#2/PM	DL:LB#2/LA	DL:LB#2/BC	OELD	DL:AD/L
MDHouston:pt	EGHylton	ASchwencer	MEWagner	TMNovak
9/19/83	9/19/83	9/19/83	9/20/83	9/19/83

Grand Gulf

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President
Claiborne County Board of Supervisors
Port Gibson, Mississippi 39150

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State of Mississippi
Jackson, Mississippi 39201

U. S. Environmental Protection Agency
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Atlanta, Georgia 30309

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State Board of Health
P. O. Box 1700
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. 10

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 June 14, 1983, June 23, 1983, and August 1, 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. 10, 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.

B. Add paragraphs 2.C.(47) and 2.C.(48) as one time exceptions to read as follows:

2.C.(47) - Relief Valve Functional Test

For the LOGIC SYSTEM FUNCTIONAL TEST required in Sections 4.4.2.1.2.b and 4.4.2.2.1.b and the ADS TRIP SYSTEM surveillance required in Table 3.3.3.-1, the provisions of Specification 4.0.4 are suspended for the portion of the surveillance that requires valve opening provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test. This is a one time exception granted for relief valve surveillance testing. This exception will expire upon the completion of the test.

2.C.(48) - Scram Discharge Volume Test

For the scram discharge volume OPERABILITY test required in Section 4.1.3.1.4.a, the provisions of Specification 4.0.4 are suspended provided that the surveillance requirement is performed within 72 hours after achieving a normal control rod configuration of less than or equal to 50% ROD DENSITY. This is a one time exception granted for scram discharge volume surveillance testing. This exception will expire upon the completion of the test.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: September , 1983

*SEE PREVIOUS CONCURRENCES

DL:LB#2/PM*	DL:LB#2/LA*	DL:LB#2/BC*	OELD*	DL:AD/L*	DL:DIR
MDHouston:pt	EGHylton	ASchwencer	MEWagner	TMNovak	DGEisenhut
9/19/83	9/19/83	9/19/83	9/20/83	9/19/83	9/23/83

B. Add paragraphs 2.C.(47) and 2.C.(48) as one time exceptions to read as follows:

2.C.(47) - <u>Relief Valve Functional Test</u>

For the LOGIC SYSTEM FUNCTIONAL TEST required in Sections 4.4.2.1.2.b and 4.4.2.2.1.b and the ADS TRIP SYSTEM surveillance required in Table 3.3.3.-1, the provisions of Specification 4.0.4 are suspended for the portion of the surveillance that requires valve opening provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test. This is a one time exception granted for relief valve surveillance testing. This exception will expire upon the completion of the test.

2.C.(48) - <u>Scram Discharge Volume Test</u>

For the scram discharge volume OPERABILITY test required in Section 4.1.3.1.4.a, the provisions of Specification 4.0.4 are suspended provided that the surveillance requirement is performed within 72 hours after achieving a normal control rod configuration of less than or equal to 50% ROD DENSITY. This is a one time exception granted for scram discharge volume surveillance testing. This exception will expire upon the completion of the test.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Date of Issuance: September , 1983

DL:LB#2/PM	DL:LB#2/LA	DL:LB#2/BC	OELD	DL:AD/L
MDHouston:pt	EGHylton	ASchwencer	MEWagner	TMNovak
9/19/83	9/19/83	9/19/83	9/20/83	9/19/83

with noted correction to 91A

ATTACHMENT TO LICENSE AMENDMENT NO. 10
FACILITY OPERATING LICENSE NO. NPF-13
DOCKET NO. 50-416

Replace the following page of the Appendix "A" Technical Specifications with enclosed page. This revised page is identified by Amendment number and contains a vertical line indicating the area of change.

REMOVE	INSERT
3/4 3-26	3/4 3-26
3/4 3-32	3/4 3-32
3/4 3-33	3/4 3-33
3/4 6-30	3/4 6-30
3/4 6-32	3/4 6-32
3/4 6-38	3/4 6-38
3/4 6-40	3/4 6-40
B3/4 3-7	B3/4 3-7

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/system	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

(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.
(e) One out-of-two taken.
* 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.
\# 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.

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**

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) Manual initiation test shall include verification of the OPERABILITY of the LPCS and LPCI injection valve interlocks. (See Note 1)

(d) This calibration shall consist of the CHANNEL CALIBRATION of the LPCS and LPCI injection valve interlocks with the interlock setpoint verified to be $\leq$ 150 psig. (See Note 1)

(e) Functional Testing of Time Delay Not Required

Note 1: Until restart after the first refueling outage, the requirements of (c) and (d) above do not apply.

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER	VALVE GROUP[(a)]	MAXIMUM ISOLATION TIME (Seconds)
Containment (Continued)				
Main Steam Line Drains	B21-F016-B	19(I)	1	15
RHR Heat Exchanger "A" to LPCI	E12-F042A-A	20(I)[(c)]	5	22
RHR Heat Exchanger "A" to LPCI	E12-F028A-A	20(I)[(c)]	5	78
RHR Heat Exchanger "A" to LPCI	E12-F037A-A	20(I)[(c)]	3	63
RHR Heat Exchanger "B" to LPCI	E12-F042B-B	21(I)[(c)]	5	22
RHR Heat Exchanger "B" to LPCI	E12-F028B-B	21(I)[(c)]	5	78
RHR Heat Exchanger "B" to LPCI	E12-F037B-B	21(I)[(c)]	3	63
RHR "A" Test Line to Supp. Pool	E12-F024A-A	23(O)[(d)]	5	90
RHR "A" Test Line to Supp. Pool	E12-F011A-A	23(O)[(d)]	5	36
RHR "C" Test Line to Supp. Pool	E12-F021-B	24(O)[(d)]	5	101
HPCS Test Line	E22-F023-C	27(O)[(d)]	6B	60
RCIC Pump Suction	E51-F031-A	28(O)[(d)]	4	56
RCIC Turbine Exhaust	E51-F077-A	29(O)[(c)]	9	26
LPCS Test Line	E21-F012-A	32(O)[(d)]	5	144
Cont. Purge and Vent Air Supply	M41-F011	34(O)	7	4
Cont. Purge and Vent Air Supply	M41-F012	34(I)	7	4
Cont. Purge and and Vent Air Exh.	M41-F034	35(I)	7	4
Cont. Purge and and Vent Air Exh.	M41-F035	35(O)	7	4
Plant Service Water Return	P44-F070-B	36(I)	6A	33
Plant Service Water Return	P44-F069-A	36(O)	6A	24
Plant Service Water Supply	P44-F053-A	37(O)	6A	24
Chilled Water Supply	P71-F150	38(O)	6A	30

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER	VALVE GROUP[a]	MAXIMUM ISOLATION TIME (Seconds)
Containment (Continued)				
Comb. Gas Control Cont. Purge (Outside Air Supply)	E61-F009	65(0)	7	4
Comb. Gas Control Cont. Purge (Outside Air Supply)	E61-F010	65(I)	7	4
Purge Rad. Detector	E61-F056	66(I)	7	4
Purge Rad. Detector	E61-F057	66(0)	7	4
RHR "B" Test Line To Suppr. Pool	E12-F024B-B	67(0)[d]	5	90
RHR "B" Test Line To Suppr. Pool	E12-F011B-B	67(0)[d]	5	27
Refueling Water Transf. Pump Suction	P11-F130	69(0)[c]	6A	4
Refueling Water Transf. Pump Suction	P11-F131	69(0)[c]	6A	8
Instr. Air to ADS	P53-F003-A	70(0)	6A	4
RCIC Turbine Exh. Vacuum Breaker	E51-F078-B	75(0)	9	7
RWCU to Feedwater	G33-F040-B	83(I)	8	30
RWCU to Feedwater	G33-F039-A	83(0)	8	29
Chemical Waste Sump Discharge	P45-F098	84(I)	6A	4
Chemical Waste Sump Discharge	P45-F099	84(0)	6A	4
Supp. Pool Clean-up Return	P60-F009-A	85(0)	6A	8
Supp. Pool Clean-up Return	P60-F010-B	85(0)	6A	4
Demin. Water Supply to Cont.	P21-F017-A	86(0)	6A	10
Demin. Water Supply to Cont.	P21-F018-B	86(I)	6A	10
RWCU Pump Suction	G33-F001-B	87(I)	8	30

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER
Containment (Continued)		
RHR Heat Ex. "C" to LPCI	E12-F234	22(0)(c)
RHR Pump "C" to LPCI	E12-F041C-B	22(I)(c)
RHR Pump "A" Test Line to Suppr. Pool	E12-F259	23(0)(e)
RHR Pump "A" Test Line to Suppr. Pool	E12-F261	23(0)(e)
RHR Pump "A" Test Line to Suppr. Pool	E12-F227	23(0)(e)
RHR Pump "A" Test Line to Suppr. Pool	E12-F262	23(0)(e)
RHR Pump "A" Test Line to Suppr. Pool	E12-F228	23(0)(e)
RHR "A" Test Line to Supp. Pool	E12-F290A-A	23(0)(d)
RHR Pump "A" Test Line to Suppr. Pool	E12-F338	23(0)(c)
RHR Pump "A" Test Line to Suppr. Pool	E12-F339	23(0)(c)
RHR Pump "A" Test Line to Suppr. Pool	E12-F260	23(0)(e)
RHR Pump "C" Test Line to Suppr. Pool	E12-F280	24(0)(e)
RHR Pump "C" Test Line to Suppr. Pool	E12-F281	24(0)(e)
HPCS Suction	E22-F014	25(0)(d)
HPCS Discharge	E22-F005	26(I)(c)
HPCS Discharge	E22-F218	26(I)(c)
HPCS Discharge	E22-F201	26(I)(c)
HPCS Test Line	E22-F035	27(0)(d)
HPCS Test Line	E22-F302	27(0)(e)
HPCS Test Line	E22-F301	27(0)(e)
LPCS Pump Suction	E21-F031	30(0)(d)
LPCS Discharge	E21-F006	31(I)(c)
LPCS Discharge	E21-F200	31(I)(c)
LPCS Discharge	E21-F207	31(I)(c)
LPCS Test Line	E21-F217	32(0)(e)
LPCS Test Line	E21-F218	32(0)(e)

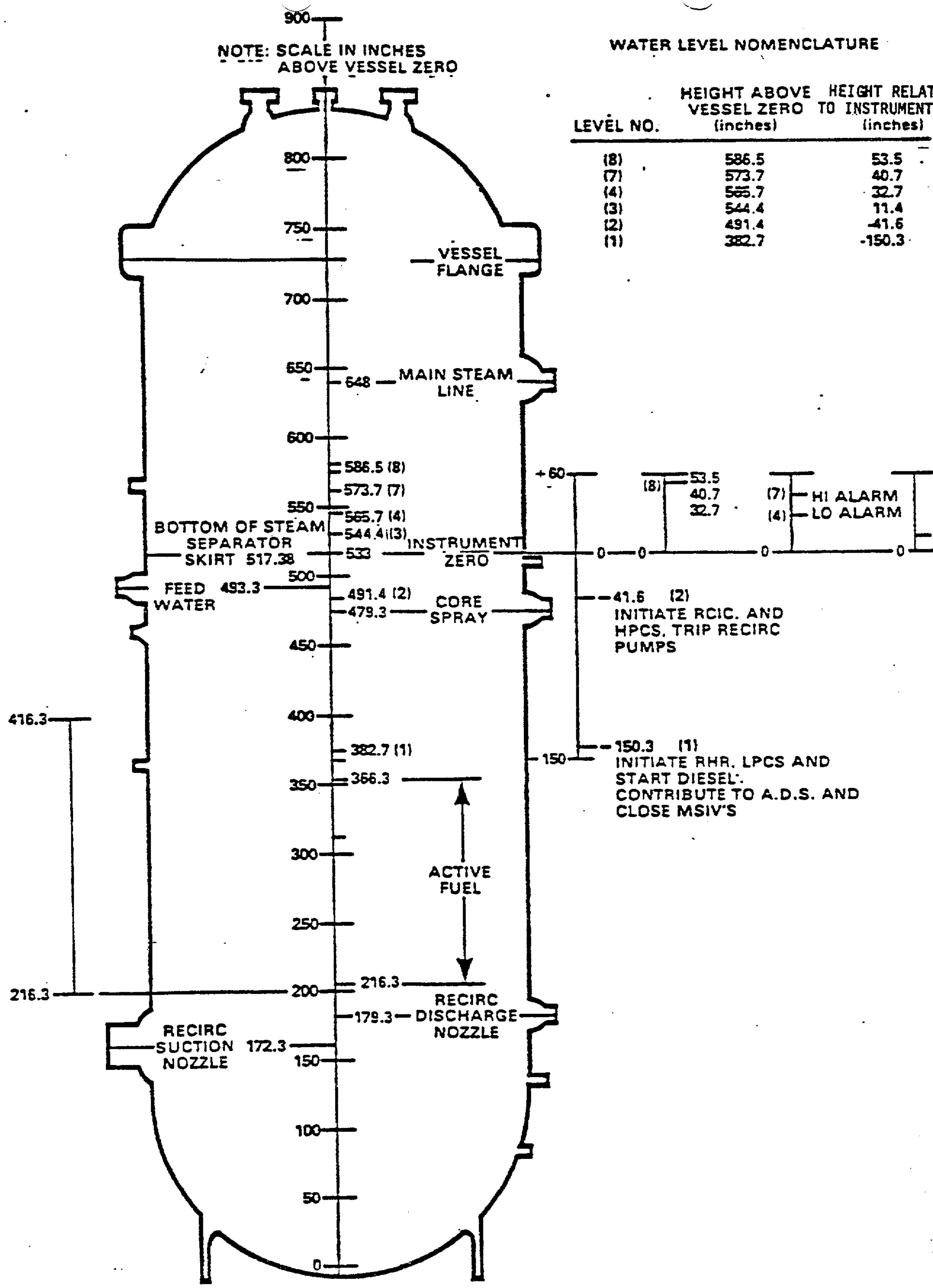
TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER
Containment (Continued)		
RHR Pump "B" Test Line	E12-F213	67(O)(e)
RHR Pump "B" Test Line	E12-F249	67(O)(e)
RHR Pump "B" Test Line	E12-F250	67(O)(e)
RHR Pump "B" Test Line	E12-F334	67(O)(c)
RHR Pump "B" Test Line	E12-E335	67(O)(c)
RHR "B" Test Line To Suppr. Pool	E12-F290B-B	67(O)(d)
Inst. Air to ADS	P53-F006	70(I)
LPCS Relief Valve Vent Header	E21-F018	71A(O)(d)
RHR Pump "C" Relief Valve Vent Header	E12-F025C	71B(O)(d)
RHR Shutdown Vent Header	E12-F036	73(O)(d)
RHR Shutdown Suction Relief Valve Disch.	E12-F005	76B(O)(d)
RHR Heat Ex. "A" Relief Vent Header	E12-F055A	77(O)(d)
RHR Heat Ex. "A" Relief Vent Header	E12-F103A	77(O)(d)
RHR Heat Ex. "A" Relief Vent Header	E12-F104A	77(O)(d)
SSW "A" Supply	P41-F169A	89(I)(c)
SSW "B" Supply	P41-F169B	92(I)(c)
Ctmt. Leak Rate Test Inst.	M61-F015	110A(I)
Ctmt. Leak Rate Test Inst.	M61-F014	110A(O)
Ctmt. Leak Rate Test Inst.	M61-F019	110C(I)
Ctmt. Leak Rate Test Inst.	M61-F018	110C(O)
Ctmt. Leak Rate Test Inst.	M61-F017	110F(I)
Ctmt. Leak Rate Test Inst.	M61-F016	110F(O)

WATER LEVEL NOMENCLATURE

LEVEL NO.	HEIGHT ABOVE VESSEL ZERO (inches)	HEIGHT RELATIVE TO INSTRUMENT ZERO (inches)
(8)	586.5	53.5
(7)	573.7	40.7
(4)	565.7	32.7
(3)	544.4	11.4
(2)	491.4	-41.6
(1)	382.7	-150.3



Bases Figure B 3/4 3-1

REACTOR VESSEL WATER LEVEL

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

SAFETY EVALUATION
AMENDMENT NO. 10 TO NPF-13
GRAND GULF NUCLEAR STATION, UNIT 1
DOCKET NO. 50-416

Introduction

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

(a) Changes to the following Technical Specifications (MP&L letters dated June 23, 1983, and August 1, 1983):

1. Tables 3.3.3-1 and 4.3.3.1-1, Bases Figure 3/4 3-1: Redefines OPERABILITY range for High Pressure Core Spray (HPCS) until first refueling outage due to water level instrumentation inaccuracies at low pressure (August 1, 1983).

2. Table 3.6.4.1: Design change to prevent automatic tripping of RHR jockey pumps, needed to prevent potential damage from waterhammer (June 23, 1983).

(b) One time exceptions to the following Technical Specifications (MP&L letters dated June 14, 1983, and August 1, 1983):

3. 4.4.2.1.2.b, 4.4.2.2.1.b and Table 3.3.3-1: Provisions of Specification 4.0.4 suspended to allow plant to attain operating conditions necessary for ADS Trip System surveillance testing (June 14, 1983).

4. 4.1.3.1.4.a: Provisions of Specification 4.0.4 suspended to allow plant to attain operating conditions necessary for Scram Discharge Volume surveillance testing (August 1, 1983).

Evaluation

(a) Technical Specification Changes

1. Tables 3.3.3-1 and 4.3.3.1-1, Bases Figure 3/4 3-1: HPCS OPERABILTY

In a letter dated August 1, 1983, MP&L requested changes to the Technical Specifications to modify the Specifications on the high pressure core spray (HPCS) system so that the injection function of Drywell Pressure - High and Manual Initiation are not required to be OPERABLE when the indicated water level on the wide range instrument is greater than Level 8 coincident with the reactor pressure being less than 1025 psig. Informal discussions between the staff and MP&L on August 25, 1983, in regard to the pressure value revised this value to 600 psig. Written confirmation of the 600 psig value was submitted by MP&L in a letter dated September 13, 1983. These changes would be in effect until modifications are made to the instrumentation but no later than startup following the first refueling outage.

The reactor water level instrumentation at Grand Gulf is the condensate chamber reference leg type. These instruments are strictly differential pressure devices which are reactor coolant density sensitive and are calibrated to be most accurate at the specific vessel conditions appropriate for the associated system functions actuated by the instrumentation. The shutdown water level range and fuel zone water level range instruments are calibrated to read accurately at atmospheric pressure; the upset, narrow and wide range water level instruments are calibrated for normal operating conditions (saturated steam at 1025 psig). At low coolant temperatures and pressures, those instruments calibrated for normal operating conditions will read higher than actual level. For example, with an actual level of 21.5" above instrument zero at 120°F and atmospheric pressure, MP&L has observed that the narrow range instrumentation would indicate a level of 32" and the wide range instruments would attempt to indicate a level of 82.5" (in actuality the upper limit of the range of the instrument is 60").

The HPCS discharge valve is interlocked closed at the vessel Level 8 setpoint (less than or equal to 55.7" above instrument zero or 220" above the active fuel). An artificially high level indication at low pressure may result in HPCS isolation when the actual vessel level is below the Level 8 setpoint. The isolation logic may be manually reset once the indicated vessel level drops below this setpoint or it will reset automatically when the indicated vessel level reaches the Level 2 HPCS initiation setpoint (a level of - 41.6" below instrument zero or 125" above the fuel).

Technical Specification Table 3.3.3-1. requires HPCS Drywell Pressure-High and Manual Initiation Actuation Instrumentation to be OPERABLE in various Operational Conditions. Actuation of these devices will result in vessel injection unless reactor vessel level is above the Level 8 setpoint or the Level 8 isolation has not been reset. The changes proposed by MP&L would add a note to Technical Specification Tables 3.3.3-1 and 4.3.3.1-1 to indicate that the injection function of Drywell Pressure-High and Manual Initiation are not required to be OPERABLE at times when a false Level 8 isolation signal is present.

Only one accident analysis presented in the Grand Gulf FSAR assumed HPCS initiation via a high drywell pressure signal. This event was a steamline break inside the drywell. MP&L presented the results of a re-analysis of this event assuming that the high drywell initiation feature was defeated and that HPCS initiated on low reactor vessel level (Level 2) only. The model accounts for density changes in the coolant by utilizing the mass of the coolant as the parameter which actually initiates HPCS injection. Thus, the mass of coolant in the vessel is the same at the time of system initiation regardless of reactor pressure. The worst single failure for the revised case was determined to be the Division I diesel generator failure. The calculated peak cladding temperature (PCT) is 1322°F versus 900°F as previously calculated without the inaccuracy in the instrumentation.

MP&L also indicated that at pressures greater than or equal to 600 psig, the indicated level error is small enough that manual reset of the HPCS isolation logic is possible for actual vessel water levels up to the high end of the normal operating range. The proposed Technical Specification, therefore, requires that the injection function be OPERABLE whenever indicated level is less than Level 8 or reactor pressure is greater than or equal to 600 psig.

Comments on these changes were received from the State of Mississippi in a letter dated August 11, 1983, and from Mr. Kenneth Lawrence by telephone on September 6, 1983. The State of Mississippi was concerned with the effect of the water level instrumentation inaccuracies on other safety analysis and with the 422°F increase in peak cladding temperature (PCT) for the reanalyzed event involving steamline break inside containment including the effect of the increased temperature upon fuel failure rate. On August 30, 1983, we discussed these concerns with Mr. Eddy Fuente, Director, Division of Radiological Health. We discussed in detail our view regarding why a predicted increase in the worst-case peak cladding temperature, an increase of 422°F to 1322°F, was not considered to be a significant safety concern. We indicated that based on data provided from tests performed at the Power Burst Facility (PBF) no significant change in the number of fuel rod failures would be expected. We also discussed that on a best-estimate basis, little or no change in the peak cladding temperature would be expected for this transient with or without HPCS initiation.

With regard to the performance of the water level instrumentation for the HPCS, we discussed with Mr. Fuente that a number of other systems would be more responsive to mitigation of accidents at low pressure and low temperature and would be available in the event of a LOCA during these infrequent modes of operation; that is during start up and shut down. The models used in the safety analysis adequately model more realistic plant situations. For licensing evaluations, analysis models compensate for density variations in the calculation of actual vessel level and PCT under LOCA conditions. The original safety analyses which initiated HPCS on high drywell pressure contained an erroneous input assumption, i.e., HPCS was always available for initiation from a high drywell pressure signal. This is not always true because HPCS lock out is possible from an indicated level signal. The indicated level signal is not density compensated at low pressure, and a false high water level signal can lock out the high drywell and manual initiation signals. The model for safety analysis itself does not use indicated levels in the calculation of PCT, but rather calculates it as a function of density. The revised analysis corrects the erroneous assumption of HPCS initiation on high drywell pressure. A mass setpoint which is independent of pressure is used to initiate HPCS. The setpoint is determined as that mass which would have a level equal to Level 2 if reactor pressure was 1000 psi. The actual level, when the liquid mass is equal to the setpoint mass, is a function of pressure. However, the actual level does not affect HPCS initiation either in the real plant or in the safety analysis model. The actual level is calculated in the model for purposes of heat transfer calculations and PCT determination. Mr. Fuente generally agreed with our conclusions and was satisfied that we had appropriately considered their comments provided in the August 11, 1983, letter.

Mr. Lawrence was also concerned about the 422°F increase in peak cladding temperature. In our discussion on September 6, 1983, we pointed out to him that the 422°F calculated increase in peak cladding temperature would result in a PCT of 1322°F which was significantly below the long established regulatory limit of 2200°F. As noted in the discussion on the subject above, we further conclude that if such a 422°F increase were to occur, it would not result in any significant change in the number of fuel failures.

Only one accident analysis was involved that assumed HPCS initiation during a time affected by a false water level reading. As to the one affected reanalysis, the delay in HPCS initiation resulted in a PCT of 1322°F. The applicable acceptance criterion for PCT, as given in 10 CFR 50.46, reads as follows: "The calculated maximum fuel element cladding temperature shall not exceed 2200°F." Since 1322°F is well below the established safety limit, these changes in the Technical Specifications have been determined not to result in exceeding regulatory limits.

The Commission has provided guidance for the application of the standards for making a "no significant hazards consideration" determination by providing examples of amendments that are considered not likely to involve significant hazards consideration (48 FR 14870). One of the examples of an amendment which will likely be found to involve no significant hazards consideration is listed as follows:

(vi) A change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan.

While the changes do result in a reduction in the safety margin in that the predicted fuel cladding temperature is increased, the peak temperature is well within the limits of 10 CFR 50.46. For this accident, a margin of over 800°F exists between the revised accident PCT of 1322°F and the 2200°F PCT limit permitted by the acceptance criteria for Emergency Core Cooling Systems for Light Water Reactors. Thus, these changes are similar to those considered in the Commission's example (vi) of changes not likely to involve significant hazards considerations.

We have reviewed the Technical Specification changes proposed by MP&L and the results of the supporting analyses. We find the changes acceptable on the basis of technical and safety considerations. We further conclude that the proposed action does not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of an accident of a type different from any evaluated previously; or

(3) Involve a significant reduction in a margin of safety.

The proposed Technical Specification changes are acceptable and since the action conforms to example (vi) given above, do not involve a significant hazards consideration.

(2) Table 3.6.4.1: Jockey Pump Isolation

By letter dated June 23, 1983, MP&L proposed the implementation of a design change package (DCP) and an accompanying change to Technical Specification Table 3.6.4.1 to correct a design deficiency reported by MP&L letter dated April 15, 1983. The design deficiency concerns the inability due to loss of the RHR jockey pumps to maintain a continuous pressurized water supply to keep the main RHR pump discharge piping full. The proposed design change will prevent the automatic tripping of the jockey pumps, partial draining of the RHR pump discharge piping and, subsequent waterhammer in the RHR system due to minimum flow valve closure.

The NRC staff has reviewed the design change which revises the logic for both the jockey pumps and their associated minimum flow valves and concludes that it is both appropriate and necessary to minimize potential waterhammer in the RHR system.

The design change will assure that a pressurized source of water is provided to the RHR pump and discharge piping continuously by removing the containment isolation signal from the minimum flow valves. Effective containment isolation for the minimum flow lines will be maintained with the jockey pump running because the pump discharge pressure is higher than containment pressure, assuring in-leakage, not out-leakage. On the other hand, when the pump is not running, the revised logic will sense this and, only in this mode, close the minimum flow valve.

Waterhammer events in nuclear power reactors have been reported and several incidents have resulted in piping and valve damage. Thus, since the changes clearly correct a deficiency in the RHR system, this action is necessary to prevent the possibility of damage to the RHR system by waterhammer, and we, therefore, conclude that the proposed action does not involve a significant increase in the probability or consequences of an accident previously evaluated or involve a significant reduction in a margin or safety. Furthermore, since the revised logic has no real effect on containment isolation, this action does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Thus, we find that the Technical Specification changes are acceptable on the basis of technical and safety considerations and that the proposed action does not involve a significant hazards consideration.

(b) One Time Technical Specification Exceptions

(3) 4.4.2.1.2.b, 4.4.2.2.1.b and Table 3.3.3-1: ADS Trip System Testing

In a letter dated June 14, 1983, MP&L requested that the provisions of Specification 4.0.4 be suspended to allow surveillance testing of safety relief valves in the ADS System after reactor steam pressure is adequate to perform the test. Specification 4.0.4 ensures that surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an applicable OPERATIONAL CONDITION or other specified applicability condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation. Suspension of the provisions of this specification will allow the plant to restart prior to performing the specified surveillance testing for the ADS Trip System. For performing this surveillance test, a reactor pressure of at least 100 psig is required. In this exception, the surveillance test must be completed within 12 hours after attaining a pressure sufficient to perform the test. Due to the low reactor pressure and temperature required to perform this test during the initial heatup with essentially unirradiated fuel, we conclude that the proposed action is acceptable on the basis of technical and safety considerations and will not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin or safety.

Thus, we find that this action is acceptable and does not involve a significant hazards consideration.

(4) 4.1.3.1.4.1: Scram Discharge Volume Testing

In a letter dated August 1, 1983, MP&L requested that the provisions of Specification 4.0.4 be suspended to allow surveillance testing of the Scram Discharge Volume after achieving the control rod configuration required to perform the test. Specification 4.0.4 ensures that surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an applicable OPERATIONAL CONDITION or other specified applicability condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation. Suspension of the provisions of this specification will allow the plant to restart prior to performing the specified surveillance testing for the Scram Discharge Volume.

For performing this surveillance test, a normal control rod configuration of less than or equal to 50% ROD DENSITY is required. In this exception, the surveillance test must be completed within 72 hours after attaining a ROD DENSITY sufficient to perform the test. Due to the low power (5-10%) required to achieve the necessary ROD DENSITY and the unirradiated condition of the initial core, we conclude that the proposed action is acceptable on the basis of technical and safety considerations and will not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

Thus, we find that this action is acceptable and does not involve a significant hazards consideration.

Environmental Consideration

We have determined that this amendment does not authorize a change in effluent types or total amount nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that this amendment involves action which is insignificant from the standpoint of environmental impact, and pursuant to CFR Section 51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this statement.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered, does not create the possibility of a new or different kind of accident from any previously evaluated, and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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:

*See previous concurrences

DL:LB#2/PM *	DL:LB#2/LA	DL:LB#2/BC *	OELD*	DL:AD/L*
MDHouston:pt	EGHylton	ASchwencer	MEWagner	TMNovak
9/19/83	9/21/83	9/19/83	9/22/83	9/19/83

(3) Involve a significant reduction in a margin of safety.

Thus, we find that this action is acceptable and does not involve a significant hazards consideration.

Environmental Consideration

We have determined that this amendment does not authorize a change in effluent types or total amount nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that this amendment involves action which is insignificant from the standpoint of environmental impact, and pursuant to CFR Section 51.5(d)(4), that an environmental impact statement or negative declartion and environmental impact appraisal need not be prepared in connection with the issuance of this statement.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered, does not create the possibility of a new or different kind of accident from any previously evaluated, and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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:

* See Previous Concurrences

DL:LB#2/PM *	DL:LB#2/LA	DL:LB#2/BC*	OELD	DL:AD/L*
MDHouston:pt	EGHylton	ASchwencer	MEWagner	TMNovak
9/19/83	9/ /83	9/19/83	9/22/83	9/19/83

For performing this surveillance test, a normal control rod configuration of less than or equal to 50% ROD DENSITY is required. In this exception, the surveillance test must be completed within 72 hours after attaining a ROD DENSITY sufficient to perform the test. Due to the low power (5-10%) required to achieve the necessary ROD DENSITY and the unirradiated condition of the initial core, we conclude that the proposed action will not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

Thus, we find that this action does not involve a significant hazards consideration.

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) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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: September , 1983

w. noted changes on pp. 4, 5 & 7

DL:LB#2/PM	DL:LB#2/BC	OELD	DL:AD/L
MDHouston:pt	ASchwencer	MWagner	TMNovak
9/19/83	9/19/83	9/20/83	9/19/83