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

February 18, 1994

Docket No. 50-458

Entergy Operations, Inc.
River Bend Station
ATTN: Mr. John R. McGaha, Jr.
Vice President - Operations
Post Office Box 220
St. Francisville, Louisiana 70775

Dear Mr. McGaha:

SUBJECT: RIVER BEND STATION, UNIT 1 - AMENDMENT NO. 72 TO FACILITY OPERATING LICENSE NO. NPF-47 (TAC NO. M88346)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 72 to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1. The amendment consists of revisions to the Technical Specifications (TS) in response to your application dated December 8, 1993 (RBG-39552), as supplemented by letter dated February 3, 1994 (RBG-400060).

The amendment permits extending the time to perform surveillance testing of certain instrumentation, valves, and electrical equipment so that the testing can be performed during the upcoming refueling outage, rather than requiring an earlier shutdown solely to perform the testing.

Your letter dated February 3, 1994, requested early issuance of this amendment because insufficient time exists for the Commission's usual 30-day notice without River Bend being required to shut down due to inability to perform the surveillance procedures with the unit in operation. The staff decided to proceed with the processing of your request in accordance with 10 CFR 50.91(a)(2), allowing the normal 30-day comment period, and issued a Notice of Enforcement Discretion on February 15, 1994, to address those surveillance requirements which became overdue on February 16 and 17, 1994.

070029

OP-1

DF01

cc w/enclosures:
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A copy of our Safety Evaluation is enclosed. The Notice of Issuance and Final Determination of No Significant Hazards Consideration and Opportunity for Hearing will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original Signed By

Robert G. Shaaf, Acting Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 72 to NPF-47
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION:

Docket File	OGC
NRC PDR	JGagliardo, RIV
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RSchaaf (2)	
CBerlinger	
DHagan	
GHill (4)	
CGrimes	
ACRS (10)	
OC/LFMB	

*For previous concurrences see attached ORC

OFFICE	PDIV-2/LA	PDIV-2/PM	PDIV-2/D	NRR:HICB	NRR:EELB
NAME	EPeyton	RSchaaf	SBlack	JWermiel	CBerlinger
DATE	2/15/94	2/15/94	2/17/94	2/16/94	2/16/94

OFFICE	OGC*				
NAME	APHodgdon				
DATE	2/10/94			/ /	/ /

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

GULF STATES UTILITIES COMPANY**

CAJUN ELECTRIC POWER COOPERATIVE AND

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-458

RIVER BEND STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 72
License No. NPF-47

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

   A. The application for amendment by Gulf States Utilities* (the licensee) dated December 8, 1993, as supplemented by letter dated February 3, 1994, 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, as amended, 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 license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

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* EOI is authorized to act as agent for Gulf States Utilities Company, which has been authorized to act as agent for Cajun Electric Power Cooperative, and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

**Gulf States Utilities Company, which owns a 70 percent undivided interest in River Bend, has merged with a wholly owned subsidiary of Entergy Corporation. Gulf States Utilities Company was the surviving company in the merger.

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment; and Paragraph 2.C.(2) of Facility Operating License No. NPF-47 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 72 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. EOI shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne C. Black

Suzanne C. Black, Director
Project Directorate IV-2
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 18, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 72

FACILITY OPERATING LICENSE NO. NPF-47

DOCKET NO. 50-458

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. The overleaf pages are provided to maintain document completeness.

REMOVE	INSERT
3/4 3-1	3/4 3-1
3/4 3-6	3/4 3-6
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
3/4 3-9	3/4 3-9
3/4 3-11	3/4 3-11
3/4 3-24	3/4 3-24
3/4 3-25	3/4 3-25
3/4 3-26	3/4 3-26
3/4 3-27	3/4 3-27
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30
3/4 3-30a	3/4 3-30a
3/4 3-40	3/4 3-40
3/4 3-41	3/4 3-41
3/4 3-42	3/4 3-42
3/4 3-43	3/4 3-43
3/4 3-59	3/4 3-59
3/4 3-63	3/4 3-63
3/4 3-64	3/4 3-64
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3/4 3-111	3/4 3-111
3/4 6-31	3/4 6-31
3/4 6-33	3/4 6-33
3/4 6-34	3/4 6-34
3/4 6-35	3/4 6-35
3/4 6-36	3/4 6-36
3/4 6-37	3/4 6-37
3/4 6-47	3/4 6-47
3/4 8-15	3/4 8-15
3/4 8-16	3/4 8-16
3/4 8-18	3/4 8-18
3/4 8-35	3/4 8-35

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within one hour.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.#

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

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2## shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.

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* An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

** The trip system need not be placed in the tripped condition if this would cause the Trip Function to occur. When a trip system can be placed in the tripped condition without causing the Trip Function to occur, place the trip system with the most inoperable channels in the tripped condition; if both systems have the same number of inoperable channels, place either trip system in the tripped condition. The requirement to place a trip system in the tripped condition does not apply to Functional Units 6 and 10 of Table 3.3.1-1.

***Logic System Functional Test period may be extended as identified by note 'p' on Table 4.3.1.1-1.

\# Channel Calibration period may be extended as identified by notes 'o' and 'q' on Table 4.3.1.1-1.

\#\# Response Time test period may be extended as identified by note '##' on Table 3.3.1-2.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors:			
a. Neutron Flux - High	2 3, 4 5(b)	3 3 3	1 2 3
b. Inoperative	2 3, 4 5	3 3 3	1 2 3
2. Average Power Range Monitor (c):			
a. Neutron Flux - High, Setdown	2 3, 4 5(b)	3 3 3	1 2 3
b. Flow Biased Simulated Thermal Power - High	1	3	4
c. Neutron Flux - High	1	3	4
d. Inoperative	1, 2 3, 4 5	3 3 3	1 2 3
3. Reactor Vessel Steam Dome Pressure - High	1, 2(d)	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Reactor Vessel Water Level-High, Level 8	1(e)	2	4
6. Main Steam Line Isolation Valve - Closure	1(e)	4	10
7. Main Steam Line Radiation - High	1, 2(d)	2	5
8. Drywell Pressure - High	1, 2(f)	2	1

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) 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 OPERABLE channel in the same trip system is monitoring that parameter.

(b) Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, the "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn*.

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 11 LPRM inputs to an APRM channel.

(d) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(e) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(f) This function is not required to be OPERABLE when DRYWELL INTEGRITY is not required.

(g) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(h) This function shall be automatically bypassed when turbine first stage pressure is $\leq$ 187 psig,** equivalent to THERMAL POWER less than 40% of RATED THERMAL POWER.

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*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.
**To allow for instrumentation accuracy, calibration and drift, a setpoint of $\leq$ 177 psig turbine first stage pressure shall be used.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - High, Setdown	NA
b. Flow Biased Simulated Thermal Power - High	≤0.09**##
c. Neutron Flux - High	≤0.09##
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	≤0.35##
4. Reactor Vessel Water Level - Low, Level 3	≤1.05
5. Reactor Vessel Water Level - High, Level 8	≤1.05
6. Main Steam Line Isolation Valve - Closure	≤0.09
7. Main Steam Line Radiation - High	NA
8. Drywell Pressure - High	NA
9. Scram Discharge Volume Water Level - High	
a. Level Transmitter	NA
b. Float Switches	NA
10. Turbine Stop Valve - Closure	≤0.06
11. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	≤0.07#
12. Reactor Mode Switch Shutdown Position	NA
13. Manual Scram	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.

**Not including simulated thermal power time constant specified in the COLR.

#Measured from start of turbine control valve fast closure.

##Response Time testing may be performed during the fifth refueling outage scheduled to begin April 16, 1994.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U,S,(b)	S/U(c), W	R	2
	S	W	R	3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor:(f)				
a. Neutron Flux - High, Setdown	S/U,S,(b)	S/U(c), W	SA	2
	S	W	SA	3, 4, 5
b. Flow Biased Simulated Thermal Power - High	S,D(h)	S/U(c), W	W(d)(e),SA(o)(q),R(i)(q)	1
c. Neutron Flux - High	S	S/U(c), W	W(d), SA	1
d. Inoperative	NA	W	NA	1, 2, 3, 4, 5
3. Reactor Vessel Steam Dome Pressure - High	S	M	R(g)(p)(q)	1, 2(j)
4. Reactor Vessel Water Level - Low, Level 3	S	M	R(g)	1, 2
5. Reactor Vessel Water Level - High, Level 8	S	M	R(g)	1
6. Main Steam Line Isolation Valve - Closure	NA	M	R	1
7. Main Steam Line Radiation - High	S	M	R	1, 2(j)
8. Drywell Pressure - High	S	M	R(g)	1, 2(l)

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
9. Scram Discharge Volume Water Level - High				
a. Level Transmitter	S	M	R(g)(p)	1, 2, 5(k)
b. Float Switch	NA	Q	R	1, 2, 5(k)
10. Turbine Stop Valve - Closure	S(m)	M(n)	R(g)(n)	1
11. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	S(m)	M(n)	R(g)(n)	1
12. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
13. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decade during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 decade during each controlled shutdown, if not performed within the previous 7 days.
(c) Within 24 hours prior to startup, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

(g) Calibrate Rosemount trip unit setpoint at least once per 31 days.

(h) Verify measured drive flow to be less than or equal to established drive flow at the existing flow control valve position.

(i) This calibration shall consist of verifying the simulated thermal power time constant is within the limits specified in the COLR.

(j) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(k) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(l) This function is not required to be OPERABLE when DRYWELL INTEGRITY is not required per Specification 3.10.1

(m) Verify the Turbine Bypass Valves are closed when THERMAL POWER is greater than or equal to 40% RATED THERMAL POWER.

(n) The CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION shall include the turbine first stage pressure instruments.

(o) The CHANNEL CALIBRATION shall exclude the flow reference transmitters; these transmitters shall be calibrated at least once per 18 months, except that this test may be performed during the fifth refueling outage scheduled to begin April 16, 1994.

(p) This period may be extended to the completion of the fifth refueling outage scheduled to begin April 16, 1994.

(q) CHANNEL CALIBRATION may be performed during the fifth refueling outage scheduled to begin April 16, 1994.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2 and with ISOLATION SYSTEM RESPONSE TIME as shown in Table 3.3.2-3.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

a. With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

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

c. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.2-1.

*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken.

**The trip system need not be placed in the tripped condition if this would cause the Trip Function to occur. When a trip system can be placed in the tripped condition without causing the Trip Function to occur, place the trip system with the most inoperable channels in the tripped condition; if both systems have the same number of inoperable channels, place either trip system in the tripped condition.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each isolation actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.2.1-1.**

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

4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation trip function shown in Table 3.3.2-3 shall be demonstrated to be within its limit at least once per 18 months.*** Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months,*** where N is the total number of redundant channels in a specific isolation trip system.

* Logic System Functional Testing period may be extended as identified by note c on Table 4.3.2.1-1.

** Channel Calibration period may be extended as identified by note 'd' on Table 4.3.2.1-1.

***Response Time test period may be extended as identified by note 'c' on Table 3.3.2-3.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL***	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level- Low Low, Level 2	1, 7, 8, 9(b)(c)(j), 15, 16	2	1, 2, 3	20
b. Drywell Pressure - High	1, 3, 8(b)(c)(j)	2	1, 2, 3	20
c. Containment Purge Isolation Radiation - High	8	1	1, 2, 3	21
2. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level- Low Low Low, Level 1	6	2	1, 2, 3	20
b. Main Steam Line Radiation - High	6, 9(d)	2	1, 2, 3	23
c. Main Steam Line Pressure - Low	6	2	1	24
d. Main Steam Line Flow - High	6	2/MSL	1, 2, 3	23
e. Condenser Vacuum - Low	6	2	1, 2**, 3**	23
f. Main Steam Line Tunnel Temperature - High	6	2	1, 2, 3	23
g. Main Steam Line Tunnel Δ Temperature - High	6	2	1, 2, 3	23
h. Main Steam Line Area Temperature High (Turbine Building)	6	2/area	1, 2, 3	23

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
6. RHR SYSTEM ISOLATION (Cont'd)		
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	$\leq$ 135 psig	$\leq$ 150 psig
f. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
7. MANUAL INITIATION	NA	NA

* See Bases Figure B 3/4 3-1.

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low Low Level 2	$\leq 10^{(a)(c)}$
b. Drywell Pressure - High	$\leq 10^{(a)(c)}$
c. Containment Purge Isolation Radiation - High(b)	NA
2. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level - Low Low Low Level 1	≤ 1.0 */$\leq 10^{(a)}$** (c)
b. Main Steam Line Radiation - High(b)	≤ 1.0 */$\leq 10^{(a)}$** (c)
c. Main Steam Line Pressure - Low	≤ 1.0 */$\leq 10^{(a)}$** (c)
d. Main Steam Line Flow - High	≤ 0.5 */$\leq 10^{(a)}$** (c)
e. Condenser Vacuum - Low	NA
f. Main Steam Line Tunnel Temperature - High	NA
g. Main Steam Line Tunnel Δ Temperature - High	NA
h. Main Steam Line Area Temperature - High (Turbine Bldg)	NA
3. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low Low Level 2	$\leq 10^{(a)(c)}$
b. Drywell Pressure - High	$\leq 10^{(a)(c)}$
c. Fuel Building Ventilation Exhaust Radiation - High(b)	NA
d. Reactor Building Annulus Ventilation Exhaust Radiation - High(b)	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. Δ Flow - High	$\leq 10^{(a)}$## (c)
b. Δ Flow Timer	NA
c. Equipment Area Temperature - High	NA
d. Equipment Area Δ Temperature - High	NA
e. Reactor Vessel Water Level - Low Low Level 2	$\leq 10^{(a)(c)}$
f. Main Steam Line Tunnel Ambient Temperature - High	NA
g. Main Steam Line Tunnel Δ Temperature - High	NA
h. SLCS Initiation	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Flow - High	$\leq 10^{(a)}$###
b. RCIC Steam Line Flow-High Timer	NA
c. RCIC Steam Supply Pressure - Low	$\leq 10^{(a)}$
d. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
e. RCIC Equipment Room Ambient Temperature - High	NA
f. RCIC Equipment Room Δ Temperature - High	NA
g. Main Steam Line Tunnel Ambient Temperature - High	NA
h. Main Steam Line Tunnel Δ Temperature - High	NA

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
i. Main Steam Line Tunnel Temperature Timer	NA
j. RHR Equipment Room Ambient Temperature - High	NA
k. RHR Equipment Room Δ Temperature - High	NA
l. RHR/RCIC Steam Line Flow - High	NA
m. Drywell Pressure - High	NA
n. Manual Initiation	NA
6. RHR SYSTEM ISOLATION	
a. RHR Equipment Area Ambient Temperature - High	NA
b. RHR Equipment Area Δ Temperature - High	NA
c. Reactor Vessel Water Level - Low Level 3	≤ 10(a)
d. Reactor Vessel Water Level - Low Low Low Level 1	≤ 10(a)(c)
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA
f. Drywell Pressure - High	NA
7. MANUAL INITIATION	NA

(a) Isolation system instrumentation response time specified includes the diesel generator starting and sequence loading delays.

(b) Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

(c) Response Time testing may be performed during the fifth refueling outage scheduled to begin April 16, 1994.

* Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed.

** Isolation system instrumentation response time for associated valves except MSIVs.

\# Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to isolation time shown in Tables 3.6.4-1 and 3.6.5.3-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

\#\# Time delay of 45-47 seconds.

\#\#\# Time delay of 3-13 seconds.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low Level 2	S	M	$R^{(b)}$	1, 2, 3
b. Drywell Pressure - High	S	M	$R^{(b)}$	1, 2, 3
c. Containment Purge Isolation Radiation - High	S	M	R	1, 2, 3
2. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level - Low Low Low Level 1	S	M	$R^{(b)}$	1, 2, 3
b. Main Steam Line Radiation - High	S	M	R	1, 2, 3
c. Main Steam Line Pressure - Low	S	M	$R^{(b)}$	1
d. Main Steam Line Flow - High	S	M	$R^{(b)}$	1, 2, 3
e. Condenser Vacuum - Low	S	M	$R^{(b)}$	1, 2**, 3**
f. Main Steam Line Tunnel Temperature - High	S	M	R	1, 2, 3
g. Main Steam Line Tunnel Δ Temperature - High	S	M	R	1, 2, 3
h. Main Steam Line Area Temperature-High (Turbine Building)	S	M	$R^{(b)}$	1, 2, 3

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
3. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low Level 2	S	M	R[b]	1, 2, 3
b. Drywell Pressure - High	S	M	R[b]	1, 2, 3
c. Fuel Building Ventilation Exhaust Radiation - High	S	M	R	*
d. Reactor Building Annulus Ventilation Exhaust Radiation - High	S	M	R	1, 2, 3
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	S	M	R[c]	1, 2, 3
b. Δ Flow Timer	NA	M	Q[c]	1, 2, 3
c. Equipment Area Temperature - High	S	M	R[c]	1, 2, 3
d. Equipment Area Δ Temperature - High	S	M	R[c]	1, 2, 3
e. Reactor Vessel Water Level - Low Low Level 2	S	M	R[b][c]	1, 2, 3
f. Main Steam Line Tunnel Ambient Temperature - High	S	M	R[c]	1, 2, 3
g. Main Steam Line Tunnel Δ Temperature - High	S	M	R[c]	1, 2, 3
h. SLCS Initiation	NA	M[a]	NA[c]	1, 2, 3

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
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	S	M	$R^{(b)}$	1, 2, 3
b. RCIC Steam Line Flow-High Timer	NA	M	Q	1, 2, 3
c. RCIC Steam Supply Pressure - Low	S	M	$R^{(b)}$	1, 2, 3
d. RCIC Turbine Exhaust Diaphragm Pressure - High	S	M	$R^{(b)}$	1, 2, 3
e. RCIC Equipment Room Ambient Temperature - High	S	M	R	1, 2, 3
f. RCIC Equipment Room Δ Temperature - High	S	M	R	1, 2, 3
g. Main Steam Line Tunnel Ambient Temperature - High	S	M	R	1, 2, 3
h. Main Steam Line Tunnel Δ Temperature - High	S	M	R	1, 2, 3
i. Main Steam Line Tunnel Temperature Timer	NA	M	Q	1, 2, 3
j. RHR Equipment Room Ambient Temperature - High	S	M	R	1, 2, 3
k. RHR Equipment Room Δ Temperature - High	S	M	R	1, 2, 3
l. RHR/RCIC Steam Line Flow-High	S	M	$R^{(b)}$	1, 2, 3
m. Drywell Pressure-High	S	M	$R^{(b)}$	1, 2, 3
n. Manual Initiation	NA	R	NA	1, 2, 3

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				
a. RHR Equipment Area Ambient Temperature - High	S	M	R	1, 2, 3
b. RHR Equipment Area Δ Temperature - High	S	M	R	1, 2, 3
c. Reactor Vessel Water Level - Low Level 3	S	M	R(b)	1, 2, 3
d. Reactor Vessel Water Level - Low Low Low Level 1	S	M	R(b)	1, 2, 3
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	S	M	R(b)(c)(d)	1, 2, 3
f. Drywell Pressure - High	S	M	R(b)	1, 2, 3
7. MANUAL INITIATION	NA	M	NA	1, 2, 3

*When handling irradiated fuel in the Fuel Building.
**When the reactor mode switch is in Run and/or any turbine stop valve is open.
(a) Each train or logic channel shall be tested at least every other 31 days.
(b) Calibrate trip unit setpoint at least once per 31 days.
(c) May be performed during the fifth refueling outage scheduled to begin April 16, 1994.
(d) CHANNEL CALIBRATION may be performed during the fifth refueling outage scheduled to begin April 16, 1994.

INSTRUMENTATION

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3.

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1.

c. With either ADS trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status:

   1. Within 7 days, provided that the HPCS and RCIC systems are OPERABLE, or

   2. Within 72 hours, provided either the HPCS or the RCIC system is inoperable.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 100 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each ECCS actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.3.1-1.##

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

##Channel Calibration and Logic System Functional testing period may be extended as identified by note b on Table 4.3.3.1-1.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

4.3.3.3 At least once per 18 months##, the ECCS RESPONSE TIME of each ECCS trip function shown in Table 3.3.3-3 shall be demonstrated to be within the limit. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months##, where N is the total number of redundant channels in a specific ECCS trip system.

##ECCS Response time testing period may be extended as identified by note A on Table 3.3.3-3.

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 III		
a. 4.16 kv Standby Bus Undervoltage (Sustained Undervoltage)	a. 4.16 kv Basis - 3045 ± 153 volts	3045 ± 214 volts
	b. 3 ± 0.3 sec. time delay	3 ± 0.33 sec. time delay
b. 4.16 kv Standby Bus Undervoltage (Degraded Voltage)	a. 4.16 kv Basis - 3777 ± 30 volts	3777 ± 75 volts
	b. 60 ± 6 sec. time delay (w/o LOCA)	60 ± 6.6 sec. time delay
	c. 3 ± 0.3 sec. time delay (w/LOCA)	3 ± 0.33 sec. time delay

*See Bases Figure B 3/4 3-1.

**(Bottom of CST is at EL 95'1".) The levels are measured from the instrument zero level of EL 98'6".

#(Bottom of suppression pool is at EL 70'.) The levels are measured from the instrument zero level of EL 89'9".

##These are inverse time delay voltage relays or instantaneous voltage relays with a time delay. The voltages shown are the maximum that will not result in a trip. Lower voltage conditions will result in decreased trip times.

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

ECCS	RESPONSE TIME (Seconds)
1. LOW PRESSURE CORE SPRAY SYSTEM	$\leq 37^{(A)}$
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM	
a. Pumps A and B	$\leq 37^{(A)}$
b. Pump C	$\leq 37^{(A)}$
3. AUTOMATIC DEPRESSURIZATION SYSTEM	NA
4. HIGH PRESSURE CORE SPRAY SYSTEM	$\leq 27^{(A)}$
5. LOSS OF POWER	NA

(A) Response time testing may be extended to the completion of the fifth refueling outage scheduled to begin April 16, 1994.

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. DIVISION I TRIP SYSTEM				
1. RHR-A (LPCI MODE) AND LPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low Low Level 1	S	M	R[a]	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	M	R[a]	1, 2, 3
c. LPCS Pump Discharge Flow-Low	S	M	R[a]	1, 2, 3, 4*, 5*
d. Reactor Vessel Pressure-Low (LPCS/LPCI Injection Valve Permissive)	S	M	R[a]	1, 2, 3, 4*, 5*
e. LPCI Pump A Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
f. LPCI Pump A Discharge Flow-Low	S	M	R[a]	1, 2, 3, 4*, 5*
g. LPCS Pump Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
h. Manual Initiation	NA	R[b]	NA	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#				
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. LPCS Pump Discharge Pressure-High	S	M	R[a]	1, 2, 3
f. LPCI Pump A Discharge Pressure-High	S	M	R[a]	1, 2, 3
g. ADS Drywell Pressure Bypass Timer	NA	M	Q	1, 2, 3
h. ADS Manual Inhibit Switch	NA	M	NA	1, 2, 3
i. Manual Initiation	NA	R	NA	1, 2, 3

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 II TRIP SYSTEM				
1. RHR B AND C (LPCI MODE)				
a. Reactor Vessel Water Level - Low Low Low Level 1	S	M	R(a)	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	M	R(a)	1, 2, 3
c. Reactor Vessel Pressure-Low (LPCI Injection Valve Permissive)	S	M	R(a)	1, 2, 3, 4*, 5*
d. LPCI Pump B Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
e. LPCI Pump Discharge Flow-Low	S	M	R(a)	1, 2, 3, 4*, 5*
f. LPCI Pump C Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
g. Manual Initiation	NA	R(b)	NA	1, 2, 3, 4*, 5*
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. ADS Drywell Pressure Bypass Timer	NA	M	Q	1, 2, 3
g. ADS Manual Inhibit Switch	NA	M	NA	1, 2, 3
h. Manual Initiation	NA	R	NA	1, 2, 3

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
C. DIVISION III 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. Pump Discharge Pressure-High	S	M	R[a]	1, 2, 3, 4*, 5*
g. HPCS System Flow Rate-Low	S	M	R[a][b]	1, 2, 3, 4*, 5*
h. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
D. LOSS OF POWER				
1. Divisions I and II				
a. 4.16 kv Standby Bus Under-voltage (Sustained Under-voltage)	S	M	R[b]	1, 2, 3, 4**, 5**
b. 4.16 kv Standby Bus Under-voltage (Degraded Voltage)	S	M	R[b]	1, 2, 3, 4**, 5**
2. Division III				
a. 4.16 kv Standby Bus Under-voltage (Sustained Under-voltage)	S	NA	R	1, 2, 3, 4**, 5**
b. 4.16 kv Standby Bus Under-voltage (Degraded Voltage)	S	M	R	1, 2, 3, 4**, 5**

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.
* When the system is required to be OPERABLE per Specification 3.5.2.
** Required when ESF equipment is required to be OPERABLE.
(a) Calibrate trip unit setpoint at least once per 31 days.
(b) May be extended to the completion of the fifth refueling outage, scheduled to begin April 16, 1994.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.1 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation channels shown in Table 3.3.4.1-1 shall be OPERABLE with their trip setpoints set consistent with values shown in the Trip Setpoint column of Table 3.3.4.1-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With an ATWS-RPT system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.1-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel trip 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, restore the inoperable channel to OPERABLE status within 30 days or be in at least STARTUP within the next 6 hours.

c. Otherwise, restore at least one inoperable channel in each trip system to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.4.1.1 Each ATWS-RPT system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.1-1.

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

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6 The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.#

#Channel Calibration period may be extended as identified by notes 'c' and 'g' on Table 4.3.6-1.

TABLE 3.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. ROD PATTERN CONTROL SYSTEM			
a. Low Power Setpoint	2	1, 2	60
b. High Power Setpoint	2	1	60
2. APRM			
a. Flow Biased Neutron Flux - Upscale	6	1	61
b. Inoperative	6	1, 2, 5	61
c. Downscale	6	1	61
d. Neutron Flux - Upscale, Startup	6	2, 5	61
3. SOURCE RANGE MONITORS			
a. Detector not full in[(a)]	3	2	61
	2**	5	62
b. Upscale[(b)]	3	2	61
	2**	5	62
c. Inoperative[(b)]	3	2	61
	2**	5	62
d. Downscale[(c)]	3	2	61
	2**	5	62
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative	6	2, 5	61
d. Downscale[(d)]	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	2	1, 2, 5*	62
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW			
a. Upscale	2	1	62

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. ROD PATTERN CONTROL SYSTEM				
a. Low Power Setpoint	$S^{(f)}$	$S/U^{(b)(e)}$, $M^{(e)}$	$SA^{\#}$	1, 2
b. High Power Setpoint	$S^{(f)}$	$S/U^{(b)(e)}$, $M^{(e)}$	$SA^{\#}$	1**
2. APRM				
a. Flow Biased Neutron Flux - Upscale	NA	$S/U^{(b)}$,M	$SA^{(g)}$	1
b. Inoperative	NA	$S/U^{(b)}$,M	NA	1, 2, 5
c. Downscale	NA	$S/U^{(b)}$,M	SA	1
d. Neutron Flux - Upscale, Startup	NA	$S/U^{(b)}$,M	SA	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	$S/U^{(b)}$,W	NA	2, 5
b. Upscale	NA	$S/U^{(b)}$,W	SA	2, 5
c. Inoperative	NA	$S/U^{(b)}$,W	NA	2, 5
d. Downscale	NA	$S/U^{(b)}$,W	SA	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	$S/U^{(b)}$,W	NA	2, 5
b. Upscale	NA	$S/U^{(b)}$,W	SA	2, 5
c. Inoperative	NA	$S/U^{(b)}$,W	NA	2, 5
d. Downscale	NA	$S/U^{(b)}$,W	SA	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	M	$R^{\#}$ (c)	1, 2, 5*
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	$S/U^{(b)}$,M	$SA^{(g)}$	1

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

b. Within 24 hours prior to startup, if not performed within the previous 7 days.

c. CHANNEL CALIBRATION may be extended to the completion of the fifth refueling outage scheduled to begin April 16, 1994.

d. [DELETED]

e. Includes reactor manual control multiplexing system input.

f. Verify the Turbine Bypass valves are closed when THERMAL POWER is greater than 20% RATED THERMAL POWER.

g. The CHANNEL CALIBRATION shall exclude the flow reference transmitters; these transmitters shall be calibrated at least once per 18 months, except that this test may be extended to the completion of the fifth refueling outage scheduled to begin April 16, 1994.

* With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

\# Calibrate trip unit setpoint once per 31 days.

** With THERMAL POWER greater than low power setpoint.

TABLE 4.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
a. Wind Speed		
1. Elev. 30 ft.	D	SA
2. Elev. 150 ft.	D	SA
b. Wind Direction		
1. Elev. 30 ft.	D	SA
2. Elev. 150 ft.	D	SA
c. Air Temperature Difference		
1. Elev. 30/150 ft.	D	SA

REMOTE SHUTDOWN MONITORING INSTRUMENTATION AND CONTROLS

LIMITING CONDITION FOR OPERATION

3.3.7.4 The remote shutdown monitoring instrumentation channels and controls shown in Table 3.3.7.4-1 and 3.3.7.4-2 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring instrumentation channels less than required by Table 3.3.7.4-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE remote shutdown system controls less than required by Table 3.3.7.4-2, restore the inoperable control(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.4.1 Each of the above required remote shutdown monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.4-1.#

4.3.7.4.2 Each of the above required remote shutdown system control circuits shall be demonstrated OPERABLE by verifying, at least once per 18 months, its capability to perform its intended function(s).

#Channel Calibration may be extended as identified by note 'a' on Table 4.3.7.4-1.

TABLE 3.3.7.4-2 (Continued)

REMOTE SHUTDOWN SYSTEM CONTROLS

	MINIMUM CHANNELS OPERABLE RSP1	RSP2
22. RHR Shutdown Cooling MOV (1E12*MOVF006A, 6B)	2[(a)]	NA
23. RHR Outboard Shutdown Isolation MOV (1E12*MOVF008)	1	NA
24. RHR Inboard Shutdown Isolation MOV (1E12*MOVF009)	1	NA
25. RHR Hx Flow to Suppression Pool MOV (1E12*MOVF011A, B)	1	1
26. RHR Reactor Head Spray MOV (1E12*MOVF023)	1	NA
27. RHR Test Line MOV (1E12*MOVF024A, B)	1	1
28. Deleted		
29. RHR Injection Shutoff MOV (1E12*MOVF027A, B)	1	1
30. RHR Upper Pool Cooling Shutoff MOV (1E12*MOVF037A, B)	1	1
31. RHR Injection MOV (1E12*MOVF042A, B, C)	1	2[(a)]
32. RHR Hx Shell Side Inlet MOV (1E12*MOVF047A, B)	1	1
33. RHR Hx Shell Side Bypass MOV (1E12*MOVF048A, B)	1	1
34. RHR Discharge to Radwaste MOV (1E12*MOVF040)	1	NA
35. Deleted		
36. RHR Injection MOV (1E12*MOVF053A, B)	1	1
37. RHR Pump Minimum Flow MOV (1E12*MOVF064A, B, C)	1	2[(a)]
38. RHR Hx Water Discharge MOV (1E12*MOVF068A, B)	1	1
39. Safety Relief Valves (1B21*RVF051C, G, D)	3[(a)]	3[(a)]
40. SSW Pump (1SWP*P2A, 2C,[(b)] 2B, 2D)	1	2[(a)]
41. Normal Service Water Isolation MOV (1SWP*MOV96A, B)	1	1
42. SSW Cooling Tower Inlet MOV (1SWP*MOV55A, B)	1	1
43. SSW Component Cooling Water Inlet MOV (1SWP*MOV510A, B)	1	1
44. SSW Component Cooling Water Outlet MOV (1SWP*MOV504A, B)	1	1

(a) One per control equipment.
(b) SSW pump 1SWP*P2C is provided on panel 1EGS*PNL4C.

TABLE 4.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Vessel Pressure	M	R[(a)]
2. Reactor Vessel Water Level	M	R[(a)]
3. Safety/Relief Valve Demand Position	M	NA
4. Suppression Pool Water Level	M	R
5. Suppression Pool Water Temperature	M	R
6. Drywell Pressure	M	R
7. Drywell Temperature	M	R
8. RHR System Flow: Loop A	M	R
Loop B	M	R
Loop C	M	R
9. RHR Hx Cooling Water System Flow: Loop A	M	R
Loop B	M	R
10. RCIC System Flow	M	R
11. RCIC Turbine Speed	M	R

(a) May be extended to be performed during the fifth refueling outage scheduled to begin April 16, 1994.

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.5 The accident monitoring instrumentation channels shown in Table 3.3.7.5-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3.7.5-1.

ACTION:

With one or more accident monitoring instrumentation channels inoperable, take the ACTION required by Table 3.3.7.5-1.

SURVEILLANCE REQUIREMENTS

4.3.7.5 Each of the above required accident monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.5-1.#

#Channel Calibration period may be extended as identified by note (a) on Table 4.3.7.5-1.

TABLE 3.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERABLE CONDITIONS	ACTION
1. Reactor Vessel Pressure	2	1	1,2,3	80
2. Reactor Vessel Water Level				
a. Wide Range	2	1	1,2,3	80
b. Fuel Zone	2	1	1,2,3	80
3. Suppression Pool Water Level	2	1	1,2,3	80
4. Suppression Pool Water Temperature	2/sector	1/sector	1,2,3	80
5. Primary Containment Pressure	2	1	1,2,3	80
6. Drywell Pressure	2	1	1,2,3	80
7. Drywell Air Temperature	2	1	1,2,3	80
8. Drywell and Primary Containment Hydrogen Concentration Analyzer and Monitor	2	1	1,2,3	80
9. Area Radiation#				
a. Primary Containment Area	2	1	1,2,3	81
b. Drywell Area	2	1	1,2,3	81

#High range gross gamma monitors.

Table 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATIONS

ACTION STATEMENTS

ACTION 80 -

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3.7.5-1, restore the inoperable channel(s) 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.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, restore the inoperable channel(s) 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 81 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

a. Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

b. Prepare and submit, within 14 days following the event, a Special Report to the Commission, pursuant to Specification 6.9.2, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	APPLICABLE OPERATIONAL CONDITIONS
1. Reactor Vessel Pressure	M	R(a)	1, 2, 3
2. Reactor Vessel Water Level			
a. Wide Range	M	R	1, 2, 3
b. Fuel Zone	M	R	1, 2, 3
3. Suppression Pool Water Level	M	R	1, 2, 3
4. Suppression Pool Water Temperature	M	R	1, 2, 3
5. Primary Containment Pressure	M	R	1, 2, 3
6. Drywell Pressure	M	R	1, 2, 3
7. Drywell Air Temperature	M	R	1, 2, 3
8. Drywell and Primary Containment Hydrogen Concentration Analyzer and Monitor	M	Q*	1, 2, 3
9. Area Radiation#			
a. Primary Containment Area	M	R** (a)	1, 2, 3
b. Drywell Area	M	R**	1, 2, 3

*Using sample gas containing:
a. One volume percent hydrogen, balance nitrogen.
b. Four volume percent hydrogen, balance nitrogen.

**The CHANNEL CALIBRATION shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr and a one point calibration check of the detector below 10 R/hr with an installed or portable gamma source.

#High range gross gamma monitors.

(a)May be extended to be performed during the fifth refueling outage scheduled to begin April 16, 1994.

3/4.3.9 PLANT SYSTEMS ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.9 The plant systems actuation instrumentation channels shown in Table 3.3.9-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.9-2.

APPLICABILITY: As shown in Table 3.3.9-1.

ACTION:

a. With a plant system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.9-2, declare the channel inoperable and take the ACTION required by Table 3.3.9-1.

b. With one or more plant systems actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.9-1.

SURVEILLANCE REQUIREMENTS

4.3.9.1 Each plant system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.9.1-1.#

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

#Channel Calibration and Logic System Functional test period may be extended as identified by note (b) on Table 4.3.9.1-1.

TABLE 3.3.9-1

PLANT SYSTEMS ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. PRIMARY CONTAINMENT VENTILATION SYSTEM - UNIT COOLER A AND B			
a. Drywell Pressure-High	2	1, 2, 3	150
b. Containment-To-Annulus ΔP High	3	1, 2, 3	151
c. Reactor Vessel Water Level-Low Low Low Level 1	2	1, 2, 3	150
d. Timers	1	1, 2, 3	152
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM			
a. Reactor Vessel Water Level-High Level 8	3	1	153

TABLE 4.3.9.1-1

PLANT SYSTEMS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT VENTILATION SYSTEM - UNIT COOLER A AND B				
a. Drywell Pressure-High	D	M	R(a)#	1, 2, 3
b. Containment-to-Annulus ΔP-High	D	M	R(a)	1, 2, 3
c. Reactor Vessel Water Level-Low Low Low Level 1	D	M	R(a)#	1, 2, 3
d. Timer	NA	M	R	1, 2, 3
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM				
a. Reactor Vessel Water Level-High Level 8	D	M	R(b)	1

(a) Calibrate trip unit setpoint once per 31 days.

(b) May be performed during the fifth refueling outage scheduled to begin April 16, 1994.

\# The specified 18 month interval during the first operation cycle may be extended to coincide with completion of the first refueling outage, scheduled to begin 9-15-87.

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP(1)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)
a. **Automatic Isolation Valves**					
1. **Primary Containment**(a) (Continued)					
RWCU Disch. to Condenser#	1G33*MOVF028	1KJB*Z4	15	20.9	Yes(f)
RWCU Return to FW#	1G33*MOVF040	1KJB*Z6	15	24.2	No
RWCU Pump Suction#	1G33*MOVF001(b)	1KJB*Z7	16	19.8	No
RWCU Pump Disch. #	1G33*MOVF053	1KJB*Z129	15	5.5	No
RWCU Disch. to Condenser#	1G33*MOVF034	1KJB*Z4	15	20.9	Yes(f)
RWCU Return to FW #	1G33*MOVF039	1KJB*Z6	15	24.2	No
RWCU Pump Suction#	1G33*MOVF004	1KJB*Z7	7	6.6	No
RWCU Pump Disch. #	1G33*MOVF054	1KJB*Z129	15	5.5	No
RWCU Backwash Disch. #	1WCS*MOV178	1KJB*Z5	1	12.1	Yes(f)
RWCU Backwash Disch. #	1WCS*MOV172	1KJB*Z5	1	12.6	Yes(f)
HPCS Test Return-Supp. Pool	1E22*MOVF023(j)	1KJB*Z11	1	50	No
RHR A Return-Supp. Pool	1E12*MOVF024A(j)	1KJB*Z24A	10	63.8	No
RHR A Hx Dump-Supp. Pool	1E12*MOVF011A(j)	1KJB*Z24A	10	34.1	No
LPCS Test Return-Supp. Pool	1E21*MOVF012(j)	1KJB*Z24A	10	57.2	No
RHR B Return-Supp. Pool	1E12*MOVF024B(j)	1KJB*Z24B	10	63.8	No
RHR B Hx Dump-Supp. Pool	1E12*MOVF011B(j)	1KJB*Z24B	10	30.8	No
RHR C Return-Supp. Pool	1E12*MOVF021(j)	1KJB*Z24C	10	97.9	No
Fuel Pool C&C Disch.	1SFC*MOV119	1KJB*Z26	1	68	No
Fuel Pool C&C Suction	1SFC*MOV120	1KJB*Z27	1	62.7	No
Fuel Pool C&C Suction	1SFC*MOV122	1KJB*Z27	1	63.8	No
Fuel Pool Purif. Suction	1SFC*MOV139	1KJB*Z28	1	39.6	No
Fuel Pool Purif. Suction	1SFC*MOV121	1KJB*Z28	1	39.6	No

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP(1)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)
a. Automatic Isolation Valves					
1. Primary Containment[a] (Continued)					
Floor Drain Disch.	1DFR*AOV102(b)	1KJB*Z35, 1DRB*Z36	1	N/A	No
Floor Drain Disch.	1DFR*AOV101[b]	1KJB*Z35, 1DRB*Z36	1	N/A	No
Equip. Drain Disch.	1DER*AOV127[b]	1KJB*Z38, 1DRB*Z39	1	N/A	No
Equip. Drain Disch.	1DER*AOV126[b]	1KJB*Z38, 1DRB*Z39	1	N/A	No
Fire Protection Hdr.	1FPW*MOV121	1KJB*Z41	1	34.1	Yes[f]
Service Air Supply	1SAS*MOV102	1KJB*Z44	1	22.0	Yes[f]
Instr. Air Supply	1IAS*MOV106	1KJB*Z46	1	18.7	Yes[f]
RPCCW Supply	1CCP*MOV138	1KJB*Z48	1	22.0	No
RPCCW Return	1CCP*MOV158	1KJB*Z49	1	23.1	No
RPCCW Return	1CCP*MOV159	1KJB*Z49	1	24.2	No
Service Water Return	1SWP*MOV5A	1KJB*Z53A	1	50.6	No
Service Water Return	1SWP*MOV5B	1KJB*Z53B	1	53.9	No
Vent. Chilled Water Rtn.	1HVN*MOV102	1KJB*Z131	1	31.9	Yes[f]
Vent. Chilled Water Rtn.	1HVN*MOV128	1KJB*Z131	1	28.6	Yes[f]
Vent. Chilled Water Sup.	1HVN*MOV127	1KJB*Z132	1	27.5	Yes[f]
Condensate Makeup Supply	1CNS*MOV125	1KJB*Z134	1	22.0	Yes[f]

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP[1]	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)
a. Automatic Isolation Valves					
1. Primary Containment[a] (Continued)					
RHR & RCIC Steam Sup.	1E51*MOVF063[b]	1KJB*Z15	2	9.9	No
RHR & RCIC Steam Sup.	1E51*MOVF076[b][m]	1KJB*Z15	2	13.4	No
RHR & RCIC Steam Sup.	1E51*MOVF064	1KJB*Z15	2	9.9	No
RCIC Pump Suc.-Supp. Pool	1E51*MOVF031[j]	1KJB*Z16	2	30.5	No
RCIC Turbine Exh.-Supp. Pool	1E51*MOVF077	1KJB*Z17	3	14.2	No
RCIC Turbine Exh. Vac. Bkrs.	1E51*MOVF078	1KJB*Z18B,C	3	16.5	No
Cont./Drywell Purge Sup.	1HVR*AOV165	1KJB*Z31	8	3	No
Cont./Drywell Purge Sup.	1HVR*AOV123	1KJB*Z31	8	3	No
Cont./Drywell Purge Outlet	1HVR*AOV128	1KJB*Z33	8	3	No
Cont./Drywell Purge Outlet	1HVR*AOV166	1KJB*Z33	8	3	No
Post-Accident Samp. Sup.	1SSR*SOV130	1KJB*Z601B	10	3	No
Post-Accident Samp. Sup.	1SSR*SOV131	1KJB*Z601B	10	3	No

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP(1)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)
a. Automatic Isolation Valves					
2. Drywell(k)					
Cont./Drywell Purge Sup.	1HVR*AOV147	1DRB*Z32	1	3	No
RPCCW Supply	1CCP*MOV142	1DRB*Z50	1	30	No
RPCCW Return	1CCP*MOV144	1DRB*Z51	1	30	No
RPCCW Return	1CCP*MOV143	1DRB*Z51	1	30	No
Service Water Supply	1SWP*MOV4A	1DRB*Z54	1	52.8	No
Service Water Supply	1SWP*MOV4B	1DRB*Z54	1	51.7	No
Service Water Return	1SWP*MOV5A	1DRB*Z55	1	50.6	No
Service Water Return	1SWP*MOV5B	1DRB*Z55	1	53.9	No
Recirc. Flow Control	1RCS*MOV58A	1DRB*Z152	1	11.0	No
Recirc. Flow Control	1RCS*MOV59A	1DRB*Z153	1	10.6	No
Recirc. Flow Control	1RCS*MOV60A	1DRB*Z154	1	6.3	No
Recirc. Flow Control	1RCS*MOV61A	1DRB*Z155	1	8.6	No
Recirc. Flow Control	1RCS*MOV58B	1DRB*Z156	1	10.6	No
Recirc. Flow Control	1RCS*MOV59B	1DRB*Z157	1	10.8	No
Recirc. Flow Control	1RCS*MOV60B	1DRB*Z158	1	6.38	No
Recirc. Flow Control	1RCS*MOV61B	1DRB*Z159	1	8.9	No
Cont./Drywell Purge Sup.	1HVR*AOV125	1DRB*Z32	1	3	No
Cont./Drywell Purge Rtn.	1HVR*AOV126	1DRB*Z34	1	3	No
Cont./Drywell Purge Rtn.	1HVR*AOV148	1DRB*Z34	1	3	No

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP(l)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)
a. Automatic Isolation Valves					
2. Drywell(k) (Continued)					
Hydrogen Mixing Line Inlet	1CPM*MOV2A	1DRB*Z57A	10	33	No
Hydrogen Mixing Line Inlet	1CPM*MOV4A	1DRB*Z57A	10	33	No
Hydrogen Mixing Line Inlet	1CPM*MOV2B	1DRB*Z57B	10	33	No
Hydrogen Mixing Line Inlet	1CPM*MOV4B	1DRB*Z57B	10	33	No
Hydrogen Mixing Line Exhaust	1CPM*MOV3A	1DRB*Z58A	10	33	No
Hydrogen Mixing Line Exhaust	1CPM*MOV1A	1DRB*Z58A	10	33	No
Hydrogen Mixing Line Exhaust	1CPM*MOV3B	1DRB*Z58B	10	33	No
Hydrogen Mixing Line	1CPM*MOV1B	1DRB*Z58B	10	33	No
Reactor Plant Sampling	1B33*AOVF019	1DRB*Z449	9	5	No
Reactor Plant Sampling	1B33*AOVF020	1DRB*Z449	9	5	No

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)
b. Manual Isolation Valves			
1. Primary Containment[a]			
LPCI A to Reactor	1E12*F099A	1KJB*Z21A	No
LPCI B to Reactor	1E12*F099B	1KJB*Z21B	No
Reactor Plant Vent. ΔP Trans.	1HVR*V8[k]	1KJB*Z602A	No
Reactor Plant Vent. ΔP Trans.	1HVR*V10[k]	1KJB*Z602B	No
PVLCS Pressure Transmitter	1LSV*V64[k]	1KJB*Z602D	No
Reactor Plant Vent. ΔP Trans.	1HVR*V12[k]	1KJB*Z602F	No
Cont. Leakage Monitor Press.	1LMS*V14	1KJB*Z603A	No
Cont. Leakage Monitor Press.	1LMS*V12	1KJB*Z603A	No
Cont. Leakage Monitor Press.	1LMS*V7	1KJB*Z603C	No
Cont. Leakage Monitor Press.	1LMS*V16	1KJB*Z603C	No
Cont. Monitor Press. Sensing	1CMS*V2[k]	1KJB*Z605A	No
Cont. Monitor Press. Sensing	1CMS*V3[k]	1KJB*Z605B	No
Reactor Plant Vent. ΔP Trans.	1HVR*V14[k]	1KJB*Z606A	No
Reactor Plant Vent. ΔP Trans.	1HVR*V16[k]	1KJB*Z606B	No
Cont. Monitor Press. Sensing	1CMS*V16[k]	1KJB*Z606C	No
Cont. Monitor Press. Sensing	1CMS*V15[k]	1KJB*Z606D	No
PVLCS Pressure Transmitter	1LSV*V65[k]	1KJB*Z606E	No
Reactor Plant Vent. ΔP Trans.	1HVR*V18[k]	1KJB*Z606F	No
LPCI A to Reactor	1E12*VF044A	1KJB*Z21A	No
LPCI B to Reactor	1E12*VF044B	1KJB*Z21B	No
SW Rtn Vacuum Release	1SWP*SOV522A[e]	1KJB*Z53A	No
SW Rtn Vacuum Release	1SWP*SOV522B[e]	1KJB*Z53B	No
SW Rtn Vacuum Release	1SWP*SOV522C[e]	1KJB*Z53A	No
SW Rtn Vacuum Release	1SWP*SOV522D[e]	1KJB*Z53B	No

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

NOTES

(a) Subject to a Type C leak rate test at a test pressure of 7.6 psig except as otherwise noted.

(b) Also isolates the drywell.

(c) Testable check valve.

(d) Isolates on MS-PLCS air line high flow or MS-PLCS air line header to Main Steam Line low differential pressure.

(e) Receives a remote manual isolation signal.

(f) This line is sealed by the penetration valve leakage control system (PVLCS). The combined leakage from valves sealed by the PVLCS is not included in 0.60 La Type B and C test total.

(g) This valve sealed by the main steam positive leakage control system (MS-PLCS). Valves sealed by the MS-PLCS are tested in accordance with Surveillance Requirement 4.6.1.3.f to verify that leakage does not exceed the limit specified in Specification 3.6.1.3.c. This leakage is not included in the 0.60 La Type B and C test total.

(h) Not subject to Type C leakage tests. Valve(s) will be included in the Type A test.

(j) Valve is hydrostatically leak tested at a test pressure of 8.36 psig (1.1 Pa). The leakage from hydrostatically tested valves is not included in the 0.60 La Type B and C test total.

(k) Not subject to a Type A, B, or C leak rate test.

(l) Valve groups listed are designated in Table 3.3.2-1.

(m) Value 1E51*MOVF076 is not required to be OPERABLE through October 4, 1986.

#The specified 18 month automatic isolation valve actuation may be performed during the fifth refueling outage scheduled to begin April 16, 1994.

3/4.6.5 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY - OPERATING

LIMITING CONDITION FOR OPERATION

3.6.5.1 SECONDARY CONTAINMENT INTEGRITY - OPERATING shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY - OPERATING restore SECONDARY CONTAINMENT INTEGRITY - OPERATING within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 SECONDARY CONTAINMENT INTEGRITY - OPERATING shall be demonstrated by:

a. Verifying at least once per 24 hours that the pressures within the Shield Building annulus, the Auxiliary Building and the Fuel Building are less than or equal to 3.0, 0.00, and 0.00 inches of vacuum water gauge, respectively.

b. Verifying at least once per 31 days that:

1. All secondary containment equipment hatch covers are installed.

2. The door in each access to the secondary containment is closed, except during normal entry and exit.

3. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers/valves secured in position.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days, and within 7 days after a battery discharge with battery terminal voltage below 110 volts or after a battery overcharge with battery terminal voltage above 144 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 at least one out of six 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-to-cell and terminal connection is less than or equal to 150×10^{-6} ohms and

   4. The battery charger will supply at least 300 amperes for chargers 1A and 1B and 50 amperes for charger 1C at a minimum of 130.2 volts for at least 8 hours.

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 the design duty cycle 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 in accordance with IEEE 450 while maintaining the battery terminal voltage greater than or equal to 105 volts.

      a) Division I
         $\geq$ 671 amperes for the first 60 seconds
         $\geq$ 270 amperes for the next 9 minutes
         $\geq$ 336 amperes for the next 60 seconds
         $\geq$ 270 amperes for the next 228 minutes
         $\geq$ 451 amperes for the last 60 seconds

#May be extended to the completion of the fifth refueling outage scheduled to begin April 16, 1994.

SURVEILLANCE REQUIREMENTS (Continued)

b) Division II
≥ 502 amperes for the first 60 seconds
≥ 261 amperes for the next 9 minutes
≥ 327 amperes for the next 60 seconds
≥ 261 amperes for the next 228 minutes
≥ 327 amperes for the last 60 seconds

c) Division III
≥ 53.2 amperes for the first 60 seconds
≥ 15.4 amperes for the next 119 minutes

e. At least once per 60 months## by verifying during shutdown 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. At least once per 18 months, during shutdown, 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.

##For Division III, may be extended to the completion of the fifth refueling outage scheduled to begin April 16, 1994.

TABLE 4.8.2.1-1

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A[1]	CATEGORY B[2]	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	>Minimum level indication mark and ≤ ¼" above maximum level indication mark	>Minimum level indication mark and ≤ ¼" above maximum level indication mark	Above top of plates and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts[c]	≥ 2.07 volts
Specific Gravity[a]	≥ 1.200[b] (Div. I&II) ≥ 1.195[b] (Div. III)	≥ 1.195 (Div. I&II) ≥ 1.190 (Div. III)	Not more than .020 below the average of all connected cells
		Average of all connected cells ≥ 1.205 (Div. I&II) ≥ 1.200 (Div. III)	Average of all connected cells ≥ 1.195[b] (Div. I&II) ≥ 1.190[b] (Div. III)

(a) Corrected for electrolyte temperature and level.
(b) Or battery charging current is less than 2 amperes when on float charge.
(c) May be corrected for average electrolyte temperature.
(1) For Category A parameters outside the limits shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameters are restored to within limits within the next 6 days.
(2) For Category B parameters outside the limits shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameters are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

ELECTRICAL POWER SYSTEMS

D.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, division I or division II and, when the HPCS system is required to be OPERABLE, division III, of the D.C. electrical power sources shall be OPERABLE with:

a. Division I consisting of:
   1. 125 volt battery 1A.
   2. 125 volt full capacity Class 1E source charger.

b. Division II consisting of:
   1. 125 volt battery 1B.
   2. 125 volt full capacity Class 1E source charger.

c. Division III consisting of:
   1. 125 volt battery 1C.
   2. 125 volt full capacity Class 1E source charger.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than the division I and/or division II battery and/or charger of the above required D.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary containment or Fuel Building, and operations with a potential for draining the reactor vessel.

b. With division III battery and/or charger of the above required D.C. electrical power sources inoperable, declare the HPCS system and the C SSW pump inoperable and take the ACTION required by Specifications 3.5.2, 3.5.3 and 3.7.1.1.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 At least the above required battery and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.#

*When handling irradiated fuel in the primary containment or Fuel Building.

#May be extended as identified by notes '#' and '##' in Surveillance Requirement 4.8.2.1.

ELECTRICAL POWER SYSTEMS

REACTOR PROTECTION SYSTEM ELECTRIC POWER MONITORING

LIMITING CONDITION FOR OPERATION

3.8.4.3 Two RPS electric power monitoring channels for each in-service RPS MG set or alternate power supply shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one RPS electric power monitoring channel for an in-service RPS MG set or alternate power supply inoperable, restore the inoperable power monitoring channel to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

b. With both RPS electric power monitoring channels for an in-service RPS MG set or alternate power supply inoperable, restore at least one electric power monitoring channel to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

SURVEILLANCE REQUIREMENTS

4.8.4.3 The above specified RPS electric power monitoring channels shall be determined OPERABLE:

a. By performance of a CHANNEL FUNCTIONAL TEST each time the unit is in COLD SHUTDOWN for a period of more than 24 hours, unless performed within the previous six months, and

b. At least once per 18 months* by demonstrating the OPERABILITY of over-voltage, under-voltage and under-frequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints.

    1. Over-voltage $\leq$ 132 VAC, Bus A and B,

    2. Under-voltage $\geq$ 115 VAC, Bus A and B, and

    3. Under-frequency 57 Hz, + 2, - 0%, Bus A and B.

*May be extended to the completion of the fifth refueling outage scheduled to begin April 16, 1994.

ELECTRICAL POWER SYSTEMS

A.C. CIRCUITS INSIDE CONTAINMENT

LIMITING CONDITION FOR OPERATION

3.8.4.4 At least the following A.C. circuits inside containment shall be de-energized*:

Equipment ID	Location	Device
1MHR*CRN1	1EJS*LDC2A	ACB022
1F42-PNLP003	1SCA-PNL8C1	Circuit Breaker 1
1F42-D002H	1SCA-PNL8C1	Circuit Breaker 15
1SFT-PNL106	1SCA-PNL8B2	Circuit Breaker 2
1SFT-PNL106	1SCA-PNL8B2	Circuit Breaker 10
1HVR*UC1AH	1SCV*PNL2A2	Circuit Breaker 5
1HVR*UC1BH	1SCV*PNL2B2	Circuit Breaker 12
1HVR-UC1CH	1SCA-PNL2C1	Circuit Breaker 9
1HVR-FN1AH	1SCA-PNL2A2	Circuit Breaker 3
1HVR-FN1BH	1SCA-PNL2F1	Circuit Breaker 6
1HVR-FN1CH	1SCA-PNL2E1	Circuit Breaker 1
1HVR-FN1DH	1SCA-PNL2B1	Circuit Breaker 6
1DRS-UC1AH	1SCA-PNL2E1	Circuit Breaker 2
1DRS-UC1BH	1SCA-PNL2F1	Circuit Breaker 3
1DRS-UC1CH	1SCA-PNL2E1	Circuit Breaker 2
1DRS-UC1DH	1SCA-PNL2F1	Circuit Breaker 3
1DRS-UC1EH	1SCA-PNL2E1	Circuit Breaker 2
1DRS-UC1FH	1SCA-PNL2F1	Circuit Breaker 3
1WCS-P5AH	1SCA-PNL2E1	Circuit Breaker 4
1WCS-P5BH	1SCA-PNL2F1	Circuit Breaker 2

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With any of the above required circuits energized, trip the associated circuit breaker(s) in the specified location within 1 hour.

SURVEILLANCE REQUIREMENTS

4.8.4.4 Each of the above required A.C. circuits shall be determined to be de-energized by verifying at least once per 24 hours** that the associated circuit breakers are in the tripped condition.

*Except during entry into the containment.
**Except at least once per 31 days if locked, sealed or otherwise secured in the tripped condition.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 72 TO FACILITY OPERATING LICENSE NO. NPF-47

ENTERGY OPERATIONS, INC.

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

1.0 INTRODUCTION

By letter dated December 8, 1993, as supplemented by letter dated February 3, 1994, Gulf States Utilities[1] (the licensee) requested an amendment to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1. The proposed amendment would revise the Technical Specifications (TS) by granting one-time extensions for certain TS surveillances which are currently required to be performed beginning February 16, 1994. The licensee is requesting extension of the surveillance intervals because the current operating cycle has been extended to approximately April 16, 1994, impacting the required completion dates for these surveillances. Performance of these surveillances within the required intervals (including the 25 percent interval extension allowed by TS 4.0.2) would require that the plant be placed in an undesirable operating configuration, or would necessitate a plant shutdown. The licensee stated that requiring the plant to shutdown solely to perform these surveillance tests would cause an unnecessary thermal transient and result in additional radiation exposure to plant personnel. The February 3, 1994, letter provided clarifying information and did not change the initial no significant hazards consideration determination.

The licensee proposed an amendment of specific TS surveillance requirements to indicate that these tests could be performed during the fifth refueling outage, scheduled to begin April 16, 1994. For certain TS requirements which remain applicable in Modes 4 and 5, the licensee proposed amending the requirements to state that the tests may be extended to the completion of the fifth refueling outage, currently scheduled for June 8, 1994. The licensee stated that these surveillance requirements required extension into the outage to support 'defense in depth' built into the outage schedule to reduce shutdown risk.

2.0 EVALUATION

Generic Letter (GL) 91-04, "Changes in Technical Specifications Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," was published April 2, 1991. The purpose of the GL was to provide guidance to licensees wishing to take advantage of improvements in reactor fuels to increase the duration of the fuel cycle for their facilities. Although the licensee is not requesting a

[1] By Amendment No. 70 to the license, effective January 1, 1994, Entergy Operations, Inc. assumed responsibility for operation of River Bend Station.

change to a 24-month fuel cycle, it is requesting a one-time surveillance extension in which some of the guidance of GL 91-04 will apply.

The staff included in its guidance in GL 91-04 the following statement:

"The NRC staff has reviewed a number of requests to extend 18-month surveillances to the end of a fuel cycle and a few requests for changes in surveillance intervals to accommodate a 24-month fuel cycle. The staff has found that the effect on safety is small because safety systems use redundant electrical and mechanical components and because licensees perform other surveillances during plant operation that confirm that these systems and components can perform their safety functions. Nevertheless, licensees should evaluate the effect on safety of an increase in 18-month surveillance intervals to accommodate a 24-month fuel cycle. This evaluation should support a conclusion that the effect on safety is small. Licensees should confirm that historical plant maintenance and surveillance data support this conclusion."

The licensee's request for surveillance extensions is very similar to one-time extensions granted previously to other NRC licensees to support extended operating cycles.

The staff has categorized the affected surveillances into four groups. The first group of surveillances includes calibration, logic system functional testing, and response time testing of certain instrumentation functions. The next group of surveillances concerns demonstration of automatic isolation of reactor water cleanup (RWCU) system containment isolation valves on receipt of an isolation test signal. The third group of surveillances concerns inspection and testing of dc batteries and battery chargers. The fourth group of surveillances concerns calibration of reactor protection system (RPS) electrical protection assemblies (EPAs). The licensee also proposed reestablishment of the baseline for the "N times 18 months" cumulative surveillance intervals for response time testing by extending the cumulative intervals to coincide with the individual extensions requested.

<u>Instrumentation Calibration, Logic System Functional Testing, and Response Time Testing Surveillance Requirements</u>

The first group of surveillances includes calibration, logic system functional testing (LSFT), and response time testing of RPS, isolation actuation system, and emergency core cooling system (ECCS) instrumentation; and calibration of control rod block, remote shutdown monitoring, accident monitoring, and feedwater system/main turbine trip system instrumentation. The licensee requested extensions of approximately 27 days to the beginning of the outage and 80 days to the end of the outage for the following instrumentation calibration surveillance requirements:

TS 4.3.1.1, RPS Instrumentation Calibration
Table 4.3.1.1-1:
- Item 2.b, "APRM Flow Biased Simulated Thermal Power - High,"

footnote (o), Flow Reference Transmitters
- Item 3, "Reactor Vessel Steam Dome Pressure - High"
- Item 9.a, "Scram Discharge Volume Water Level - High" *

TS 4.3.2.1, Isolation Actuation Instrumentation Calibration
Table 4.3.2.1-1:
- Item 6.e, "RHR System Isolation - Reactor Vessel (RHR Cut-in Permissive) Pressure - High"

TS 4.3.3.1, ECCS Actuation Instrumentation Calibration
Table 4.3.3.1-1:
- Item C.1.f, "HPCS System Pump Discharge Pressure - High" *
- Item D.1.a, "Loss of Power - 4.16kv Standby Bus Undervoltage (Sustained Undervoltage)" *
- Item D.1.b, "Loss of Power - 4.16kv Standby Bus Undervoltage (Degraded Undervoltage)" *

TS 4.3.6, Control Rod Block Instrumentation Calibration
Table 4.3.6-1:
- Item 2.a, "APRM Flow Biased Simulated Thermal Power - High," footnote (g), Flow Reference Transmitters
- Item 5.a, "Scram Discharge Volume Water Level - High" *
- Item 6.a, "Reactor Coolant System Recirculation Flow - Upscale," footnote (g), Flow Reference Transmitters

TS 4.3.7.4.1, Remote Shutdown Monitoring Instrumentation Calibration
Table 4.3.7.4-1:
- Item 1, "Reactor Vessel Pressure"
- Item 2, "Reactor Vessel Water Level"

TS 4.3.7.5, Accident Monitoring Instrumentation Calibration
Table 4.3.7.5-1:
- Item 1, "Reactor Vessel Pressure"
- Item 9.b, "Drywell Area Radiation Monitor"

TS 4.3.9.1, Plant Systems Actuation Instrumentation Calibration
Table 4.3.9.1-1:
- Item 2.a, "Feedwater System/Main Turbine Trip System - Reactor Vessel Water Level - High Level 8"

* Extension required to the end of the outage to provide 'defense in depth' during shutdown operations.

The licensee stated that observed drift characteristics, as well as the presence of redundant and diverse channels for most of the affected instrumentation, support extension of these surveillance intervals. The affected instrumentation is subject to periodic channel checks, channel functional tests, and channel calibrations which will continue to be performed during the extension period. Based on the above, and the relatively short time period of the requested extension, the staff finds the proposed calibration surveillance interval extensions acceptable.

Logic systems are comprised of detection devices activated by certain physical conditions (e.g., pressure switches, temperature switches, etc.) and decision making relay networks that will cause a safety system component or device (e.g., pump, valve, etc.) to operate when needed. Logic system functional tests are surveillance tests of all relays and contacts, trip units, solid state logic elements, and related components from sensor through actuated device to verify system operability. The licensee requested extensions of approximately 29 days to the beginning of the outage and 80 days to the end of the outage for the following LSFT surveillance requirements:

TS 4.3.1.2, RPS Instrumentation LSFT

Table 4.3.1.1-1:

- Item 3, "Reactor Vessel Steam Dome Pressure - High"
- Item 9.a, "Scram Discharge Volume Water Level - High"

TS 4.3.2.2, Isolation Actuation Instrumentation LSFT

Table 4.3.2.1-1:

Item 4, "RWCU Isolation":

- Item 4.a, "Differential Flow - High"
- Item 4.b, "Differential Flow Timer"
- Item 4.c, "Equipment Area Temperature - High"
- Item 4.d, "Equipment Area Differential Temperature - High"
- Item 4.e, "Reactor Vessel Water Level - Low Low Level 2"
- Item 4.f, "Main Steam Line Tunnel Ambient Temperature - High"
- Item 4.g, "Main Steam Line Tunnel Differential Temperature - High"
- Item 4.h, "SLCS Initiation"

Item 6, "RHR System Isolation":

- Item 6.e, "Reactor Vessel (RHR Cut-in Permissive) Pressure - High"

TS 4.3.3.2, ECCS Actuation Instrumentation LSFT

Table 4.3.3.1-1:

- Item C.1.f, "HPCS System Pump Discharge Pressure - High" *
- Item D.1.a, "Loss of Power - 4.16kv Standby Bus Undervoltage (Sustained Undervoltage)" *
- Item D.1.b, "Loss of Power - 4.16kv Standby Bus Undervoltage (Degraded Undervoltage)" *

TS 4.3.9.2, Plant Systems Actuation Instrumentation LSFT

Table 4.3.9.1-1:

- Item 2.a, "Feedwater System/Main Turbine Trip System - Reactor Vessel Water Level - High Level 8"

* Extension required to the end of the outage to provide 'defense in depth' during shutdown operations.

The licensee stated that industry reliability studies for boiling water reactors (BWRs) prepared by the BWR Owners' Group (NEDC-30936P) show that overall safety system reliability is not dominated by the logic system reliability, but by the reliability of the mechanical components (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability. Based on the above, and the relatively short time period of the requested extension, the staff finds the proposed LSFT surveillance interval extensions acceptable.

Protection system response time is the time interval from when a monitored parameter exceeds its actuation setpoint at the channel sensor to the time at which the actuated equipment reaches the required state (e.g., deenergization of RPS scram pilot solenoids, completion of valve movement to the required position, etc.). The licensee requested extensions of approximately 59 days to the beginning of the outage and 100 days to the end of the outage for the following instrumentation response time test surveillance requirements:

TS 4.3.1.1, RPS Instrumentation Calibration
Table 4.3.1.1-1:
- Item 2.b, "APRM Flow Biased Simulated Thermal Power - High," footnote (i), Simulated Thermal Power Time Constant (Calibration of this time constant is essentially a response time test.)

TS 4.3.1.3, RPS Instrumentation Response Time Test
Table 3.3.1-2:
- Item 2.b, "APRM Flow Biased Simulated Thermal Power - High"
- Item 2.c, "APRM Neutron Flux - High"
- Item 3, "Reactor Vessel Steam Dome Pressure - High"

TS 4.3.2.3, Isolation Actuation Instrumentation Response Time Test
Table 3.3.2-3:
Item 1, "Primary Containment Isolation":
- Item 1.a, "Reactor Vessel Water Level - High"
- Item 1.b, "Drywell Pressure - High"

Item 2, "Main Steam Line Isolation":
- Item 2.a, "Reactor Vessel Water Level - Low Low Low Level 1"
- Item 2.b, "Main Steam Line Radiation - High"
- Item 2.c, "Main Steam Line Pressure - Low"
- Item 2.d, "Main Steam Line Flow - High"

Item 3, "Secondary Containment Isolation":
- Item 3.a, "Reactor Vessel Water Level - Low Low Level 2"
- Item 3.b, "Drywell Pressure - High"

Item 4, "RWCU Isolation":
- Item 4.a, "Differential Flow - High"
- Item 4.e, "Reactor Vessel Water Level - Low Low Level 2"

Item 6, "RHR System Isolation":
- Item 6.d, "Reactor Vessel Water Level - Low Low Low Level 1"

TS 4.3.3.3, ECCS Actuation Instrumentation Response Time Test
Table 3.3.3-3:

- Item 1, "Low Pressure Core Spray System" *
- Item 2.a, "LPCI Mode of RHR System - Pumps A and B" *
- Item 2.b, "LPCI Mode of RHR System - Pump C" *
- Item 4, "High Pressure Core Spray System" *

* Extension required to the end of the outage to provide 'defense in depth' during shutdown operations.

The licensee stated that there are redundant and diverse channels available to perform each of the affected functions. Instrumentation response times and failure probabilities are small fractions of the overall system response times and failure probabilities. With respect to the main steam line radiation monitor response time test (TS 4.3.2.3, Table 3.3.2-3, Item 2.b), the licensee referred to a General Electric Licensing Topical Report, NEDO-31400, which justified removal of the main steam isolation valve (MSIV) closure and reactor scram functions of the main steam line radiation monitors (MSLRMs). The licensee stated that the topical report is applicable to River Bend Station. The NRC has previously reviewed and approved this topical report in a safety evaluation dated May 15, 1991. The staff determination that the subject MSLRM functions are not required for safe operation provides additional justification for extension of the subject surveillance interval. The extensions requested are for a short time period relative to the required surveillance intervals. Based on the above, the staff finds the licensee's request for a one-time extension of the response time testing surveillance intervals to be acceptable.

Cumulative Response Time Testing Surveillance Interval Baseline

The licensee also proposed reestablishment of the baseline for the "N times 18 months" cumulative surveillance interval for response time testing by extending the cumulative surveillance interval to coincide with the individual extensions discussed above. The licensee stated that extending the cumulative intervals will ensure that future response time testing intervals will not become overdue prematurely due to the interval extensions requested by this amendment. Extension of the cumulative intervals would not be for more than the individual extensions requested. Due to the fact that the individual extensions have been shown to be acceptable as discussed above, extending the cumulative surveillance interval for response time testing is acceptable to the staff.

Reactor Water Cleanup System Containment Isolation Valves

TS Surveillance Requirement 4.6.4.2 requires demonstration of automatic actuation of the isolation valves listed in Table 3.6.4-1 on receipt of an isolation test signal. The licensee requested an extension of approximately 13 days for the surveillance interval requirement for eight RWCU system valves.

The licensee stated that the containment penetrations have redundancy such that failure of a single valve does not prevent containment isolation. Furthermore, the containment isolation system is subject to periodic testing, including inservice tests of the valves. Based on the above, and the short time period of the requested extension, the staff finds the proposed isolation valve surveillance interval extensions acceptable.

DC Battery and Charger Inspection and Testing

The third group of surveillances concerns inspection (TSs 4.8.2.1.c.1, c.2, and c.3), service tests (TS 4.8.2.1.d.1), and performance tests (TS 4.8.2.1.e) of dc batteries; and load tests of the battery chargers (TS 4.8.2.1.c.4). The licensee requested extension of these surveillance interval requirements until the end of the refueling outage to provide 'defense in depth' during shutdown operations, an estimated extension period of 66 days.

The licensee stated that past battery inspections have found no visual abnormalities or unacceptable resistance measurements. Past service tests of the batteries have consistently yielded acceptable results. Pilot cells, monitored weekly, have not indicated any degraded conditions. Quarterly measurements of cell voltage, temperature, and specific gravity have not indicated any battery degradation. The licensee stated that the charger load tests have always yielded satisfactory results, with the voltage never falling below the test acceptance criteria. Finally, the extensions requested are for a short time period relative to the required surveillance intervals. Based on the above, the staff finds the licensee's request for a one-time extension of the dc battery and charger surveillance intervals to be acceptable.

The licensee also proposed an editorial change to TS surveillance requirement 4.8.2.2, which references TS 4.8.2.1, for the surveillances required to demonstrate operability of dc sources required during shutdown operations.

The licensee proposed to include a note for this surveillance requirement to provide consistency with TS 4.8.2.1 surveillance items for which extensions have been proposed. This is primarily an editorial change to maintain consistency between these specifications. The surveillance interval extensions have been found acceptable by the staff as noted above. Therefore, this change is acceptable to the staff.

RPS Electrical Protection Assemblies

The fourth group of surveillances concerns calibration of RPS electrical protection assemblies. The licensee requested an extension of these surveillance interval requirements until the end of the refueling outage, an extension of approximately 42 days.

The RPS consists of two independent trip systems; each subsystem has redundant channels. The RPS logic is such that a single failure will neither cause nor prevent a reactor scram. The licensee stated that laboratory testing of the EPAs has exhibited little or no drift and a review of the operating history of the EPAs indicated only one failure attributed to drift since 1985. Based on the observed lack of drift of the EPAs, the redundancy of the system logic, and the relatively short time period of the requested extension, the staff finds the proposed surveillance interval extension acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (59 FR 2630). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

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

Principal Contributor: R. Schaaf

Date: February 18, 1994