ATTN: Mr. John R. M[illegible]ha, Jr.
Vice Presiden[illegible] (RBNG)
Post Office Box 220
St. Francisville, Louisiana 70775

March [illegible], 1995

SUBJECT: RIVER BEND STATION, UNIT 1 - AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NO. NPF-47 (TAC NOS. M88871 AND M88872)

Dear Mr. McGaha:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 76 to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1. The amendment consists of revising the technical specifications (TSs) in response to your application dated January 14, 1994 (RBG-39894) as supplemented by letters dated November 10, 1994, and February 8, 1995.

The proposed amendment revises various TSs, removing component lists from the TSs and relocating them to the Technical Requirements Manual. The removal of the component lists is a line item improvement of the TSs, consistent with the guidance contained in Generic Letter 91-08. The amendment also relocates the reactor vessel material specimen withdrawal schedule from the TSs to the updated safety analysis report. This revision implements Generic Letter 91-01.

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

Sincerely,
ORIGINAL SIGNED BY:
David L. Wigginton, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosures: 1. Amendment No. 76 NPF-47
2. Safety Evaluation
cc w/encls: See next page

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Document Name: RB88872.AMD #95-012 #8 2/15/95

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

March 8, 1995

Entergy Operations, Inc.
ATTN: Mr. John R. McGaha, Jr.
Vice President (RBNG)
Post Office Box 220
St. Francisville, Louisiana 70775

SUBJECT: RIVER BEND STATION, UNIT 1 - AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NO. NPF-47 (TAC NOS. M88871 AND M88872)

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David L. Wigginton, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-458

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2. Safety Evaluation

cc w/encls: See next page

Entergy Operations, Inc.

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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. 76
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 January 14, 1994, as supplemented by letters dated November 10, 1994, and February 8, 1995, 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

---

* 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. 76 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

David L. Wigginton, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 8, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 76

FACILITY OPERATING LICENSE NO. NPF-47

DOCKET NO. 50-458

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

REMOVE	INSERT
xxv	xxv
xxvi	xxvi
1-6	1-6
3/4 3-1	3/4 3-1
3/4 3-7	3/4 3-7
3/4 3-9	3/4 3-9
3/4 3-11	3/4 3-11
3/4 3-12	3/4 3-12
3/4 3-13	3/4 3-13
3/4 3-14	3/4 3-14
3/4 3-15	3/4 3-15
3/4 3-16	3/4 3-16
3/4 3-18	3/4 3-18
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 3-41	3/4 3-41
3/4 3-42	3/4 3-42
3/4 3-43	3/4 3-43
3/4 3-107	3/4 3-107
3/4 3-111	3/4 3-11
3/4 4-11	3/4 4-11
3/4 4-12	3/4 4-12
3/4 4-13	3/4 4-13
3/4 4-22	3/4 4-22
3/4 4-24	3/4 4-24
3/4 6-3	3/4 6-3
3/4 6-4	3/4 6-4
3/4 6-5	3/4 6-5
3/4 6-6	3/4 6-6
3/4 6-7	3/4 6-7
3/4 6-31	3/4 6-31
3/4 6-31a	3/4 6-31a
3/4 6-32 thru 3/4 6-47	3/4 6-32
3/4 6-51	3/4 6-51
3/4 6-52	3/4 6-52

REMOVE	INSERT
3/4 6-53	3/4 6-53
3/4 6-54	3/4 6-54
3/4 8-25	3/4 8-25
3/4 8-26	3/4 8-26
3/4 8-28 thru 3/4 8-32	3/4 8-28
3/4 8-33	3/4 8-33
3/4 8-34	3/4 8-34
3/4 8-36	3/4 8-36
6-9	6-9
6-14	6-14

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

MEMBER(S) OF THE PUBLIC

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

MINIMUM CRITICAL POWER RATIO

1.26 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.27 The OFFSITE DOSE CALCULATION MANUAL shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints. It shall also contain a table and figure defining current radiological environmental monitoring sample locations.

OPERABLE - OPERABILITY

1.28 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION - CONDITION

1.29 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

1.30 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.31 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY - FUEL HANDLING

1.32 PRIMARY CONTAINMENT INTEGRITY - FUEL HANDLING shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position. Up to twelve vent and drain line pathways may be opened under administrative control for the purposes of surveillance testing provided the total calculated flow rate through the open vent and drain line pathways is less than or equal to 70.2 cfm.

b. All primary containment hatches are closed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.6.1.4.

PRIMARY CONTAINMENT INTEGRITY - OPERATING

1.33 PRIMARY CONTAINMENT INTEGRITY - OPERATING shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Specification 3.6.4.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.6.1.4.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.3.

e. The suppression pool is in compliance with the requirements of Specification 3.6.3.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM (PCP)

1.34 The PROCESS CONTROL PROGRAM shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 61, 10 CFR Part 71 and

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.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With one channel required by Table 3.3.1-1 inoperable in one or more Functional Units, place the inoperable channel and/or that Trip System in the tripped condition* within 12 hours.

b. With two or more channels required by Table 3.3.1-1 inoperable in one or more Functional Units;

1. Within one hour, verify sufficient channels remain OPERABLE or in the tripped condition* to maintain trip capability in the Functional Unit, and

2. Within six hours, place the inoperable channel(s) in one Trip System and/or that Trip System** in the tripped condition*, and

3. Within 12 hours, restore the inoperable channels in the other Trip System to an OPERABLE status or place the inoperable channels in the tripped condition*.

Otherwise, take the ACTION required by Table 3.3.1-1 for the Functional Unit.

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 required reactor trip functional unit shall be demonstrated to be within its limit at least once per 18 months. Neutron detectors are exempt from response time testing. 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.

---

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

** This Action applies to that Trip System with the most inoperable channels; if both Trip Systems have the same number of inoperable channels, the ACTION can be applied to either Trip System.

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

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 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)], Q	W[(d)(e)],SA[(o)],R[(i)]	1
c. Neutron Flux - High	S	S/U[(c)], Q	W[(d)], SA	1
d. Inoperative	NA	Q	NA	1, 2, 3, 4, 5
3. Reactor Vessel Steam Dome Pressure - High	S	Q	R[(g)]	1, 2[(j)]
4. Reactor Vessel Water Level - Low, Level 3	S	Q	R[(g)]	1, 2
5. Reactor Vessel Water Level - High, Level 8	S	Q	R[(g)]	1
6. Main Steam Line Isolation Valve - Closure	NA	Q	R	1
7. Main Steam Line Radiation - High	S	Q	R	1, 2[(j)]
8. Drywell Pressure - High	S	Q	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 IN WHICH SURVEILLANCE REQUIRED
9. Scram Discharge Volume Water Level - High				
a. Level Transmitter	S	Q	R(g)(p)	1, 2, 5(k)
b. Float Switch	NA	Q	R	1, 2, 5(k)
10. Turbine Stop Valve - Closure	S(m)	Q(n)	R(g)(n)	1
11. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	S(m)	Q(n)	R(g)(n)	1
12. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
13. Manual Scram	NA	W	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 $\geq$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 92 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.

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.

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 requirements for one Trip System,

   1. Within 12 hours for Trip Functions 1.b, 2.b, 3.b, 6.c, 6.e, and 6.f, and

   2. Within 24 hours for Trip Functions other than 1.b, 2.b, 3.b, 6.c, 6.e, and 6.f,

   place the inoperable channel(s) and /or that Trip System in the tripped condition*.

c. With the number of OPERABLE Channels less than required by the Minimum OPERABLE Channels per Trip System requirements for both Trip Systems,

   1. Within one hour, place the inoperable channel(s) in one Trip System and/or that Trip System** in the tripped condition*, and

   2. Within 12 hours for Trip Functions 1.b, 2.b, 3.b, 6.c, 6.e, and 6.f, and within 24 hours for Trip Functions other than 1.b, 2.b, 3.b, 6.c, 6.e, and 6.f, place the inoperable channel(s) in the other Trip System in the tripped condition*.

---

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

** This ACTION applies to that Trip System with the most inoperable channels; if both Trip Systems have the same number of inoperable channels, the ACTION can be applied to either Trip System.

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 required isolation trip function 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.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

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

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	APPLICABLE OPERATIONAL CONDITION	ACTION
3. SECONDARY CONTAINMENT ISOLATION			
a. Reactor Vessel Water Level-Low Low, Level 2[(b)(c)(e)(h)(i)]	2	1, 2, 3	25
b. Drywell Pressure - High[(b)(c)(e)(h)(i)]	2	1, 2, 3	25
c. Fuel Building Ventilation Exhaust Radiation - High[(e)(h)]	1	*	28
d. Reactor Building Annulus Ventilation Exhaust Radiation - High[(b)(e)(i)]	1	1, 2, 3	29
4. REACTOR WATER CLEANUP SYSTEM ISOLATION			
a. Δ Flow - High	1	1, 2, 3	27
b. Δ Flow Timer	1	1, 2, 3	27
c. Equipment Area Temperature - High	1	1, 2, 3	27
d. Equipment Area Δ Temperature - High	1	1, 2, 3	27
e. Reactor Vessel Water Level - Low Low, Level 2	2	1, 2, 3	27
f. Main Steam Line Tunnel Ambient Temperature - High	1	1, 2, 3	27

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]	APPLICABLE OPERATIONAL CONDITION	ACTION
4. REACTOR WATER CLEANUP SYSTEM ISOLATION (Continued)			
g. Main Steam Line Tunnel Δ Temperature - High	1	1, 2, 3	27
h. SLCS Initiation	1[f]	1, 2, 3	27
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION			
a. RCIC Steam Line Flow - High	1	1, 2, 3	27
b. RCIC Steam Line Flow - High Timer	1	1, 2, 3	27
c. RCIC Steam Supply Pressure - Low	1	1, 2, 3	27
d. RCIC Turbine Exhaust Diaphragm Pressure - High	2	1, 2, 3	27
e. RCIC Equipment Room Ambient Temperature - High	1	1, 2, 3	27
f. RCIC Equipment Room Δ Temperature - High	1	1, 2, 3	27
g. Main Steam Line Tunnel Ambient Temperature - High	1	1, 2, 3	27
h. Main Steam Line Tunnel Δ Temperature - High	1	1, 2, 3	27

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	APPLICABLE OPERATIONAL CONDITION	ACTION
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION (continued)			
i. Main Steam Line Tunnel Temperature Timer	1	1, 2, 3	27
j. RHR Equipment Room Ambient Temperature - High	1	1, 2, 3	27
k. RHR Equipment Room Temperature - High	1	1, 2, 3	27
l. RHR/RCIC Steam Line Flow - High	1	1, 2, 3	27
m. Drywell Pressure - High[(g)]	1	1, 2, 3	27
n. Manual Initiation[(k)]	1	1, 2, 3	26
6. RHR SYSTEM ISOLATION			
a. RHR Equipment Area Ambient Temperature - High	2	1, 2, 3	30
b. RHR Equipment Area Δ Temperature - High	2	1, 2, 3	30
c. Reactor Vessel Water Level - Low, Level 3	2	1, 2, 3	30
d. Reactor Vessel Water Level - Low Low Low, Level 1	2	1, 2, 3	30

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]	APPLICABLE OPERATIONAL CONDITION	ACTION
6. RHR SYSTEM ISOLATION (continued)			
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	2	1, 2, 3	30
f. Drywell Pressure - High	2	1, 2, 3	30
7. MANUAL INITIATION[l]	2	1, 2, 3	22

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION
ACTION

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

ACTION 21 - Close the affected system isolation valve(s) within one hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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

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

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

ACTION 25 - Within 1 hour, establish SECONDARY CONTAINMENT INTEGRITY - OPERATING with the standby gas treatment system and Fuel Building Ventilation System (emergency mode) operating.

ACTION 26 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within 1 hour and declare the affected system inoperable.

ACTION 27 - Close the affected system isolation valves within 1 hour and declare the affected system inoperable.

ACTION 28 - Within 1 hour, initiate and maintain the Fuel Building Ventilation System in the emergency mode of operation.

ACTION 29 - Within 1 hour, initiate and maintain annulus mixing system with the reactor building annulus exhaust to at least one operating standby gas treatment train.

ACTION 30 - Within 1 hour, lock the affected system isolation valves closed and declare the affected system inoperable.

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION
ACTION

NOTES

* When handling irradiated fuel in the Fuel Building.
** May be bypassed with reactor mode switch not in Run and all turbine stop valves closed.

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.
(b) Also actuates the standby gas treatment system.
(c) Also actuates the main control room air conditioning system in the emergency mode of operation.
(d) Also trips and isolates the air removal pumps.
(e) Also actuates secondary containment ventilation isolation dampers.
(f) Manual initiation of SLCS pump C001B closes 1G33*MOVF001, and manual initiation of SLCS pump C001A closes 1G33*MOVF004.
(g) Requires RCIC system steam supply pressure-low coincident with drywell pressure-high.
(h) Also starts the Fuel Building Exhaust Filter Trains A and B.
(i) Also starts the Annulus Mixing System.
(j) Also actuates the containment hydrogen analyzer/monitor recorder.
(k) Manual initiation isolates the outboard steam supply isolation valve only and only following a manual or automatic initiation of the RCIC system.
(l) Valve 1E22*MOVF023 does not isolate on the manual initiation.

Table 3.3.2-3 has been deleted.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP UNIT	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	Q	R(b)	1, 2, 3
b. Drywell Pressure - High	S	Q	R(b)	1, 2, 3
c. Containment Purge Isolation Radiation - High	S	Q	R	1, 2, 3
2. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level - Low Low Low Level 1	S	Q	R(b)	1, 2, 3
b. Main Steam Line Radiation - High	S	Q	R	1, 2, 3
c. Main Steam Line Pressure - Low	S	Q	R(b)	1
d. Main Steam Line Flow - High	S	Q	R(b)	1, 2, 3
e. Condenser Vacuum - Low	S	Q	R(b)	1, 2**, 3**
f. Main Steam Line Tunnel Temperature - High	S	Q	R	1, 2, 3
g. Main Steam Line Tunnel ΔTemperature - High	S	Q	R	1, 2, 3
h. Main Steam Line Area Temperature-High (Turbine Building)	S	Q	R(b)	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP UNIT	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	Q	$R^{(b)}$	1, 2, 3
b. RCIC Steam Line Flow-High Timer	NA	Q	Q	1, 2, 3
c. RCIC Steam Supply Pressure - Low	S	Q	$R^{(b)}$	1, 2, 3
d. RCIC Turbine Exhaust Diaphragm Pressure - High	S	Q	$R^{(b)}$	1, 2, 3
e. RCIC Equipment Room Ambient Temperature - High	S	Q	R	1, 2, 3
f. RCIC Equipment Room Δ Temperature - High	S	Q	R	1, 2, 3
g. Main Steam Line Tunnel Ambient Temperature - high	S	Q	R	1, 2, 3
h. Main Steam Line Tunnel Δ Temperature - High	S	Q	R	1, 2, 3
i. Main Steam Line Tunnel Temperature Timer	NA	Q	Q	1, 2, 3
j. RHR Equipment Room Ambient Temperature - High	S	Q	R	1, 2, 3
k. RHR Equipment Room Δ Temperature - High	S	Q	R	1, 2, 3
l. RHR/RCIC Steam Line Flow-High	S	Q	$R^{(b)}$	1, 2, 3
m. Drywell Pressure-High	S	Q	$R^{(b)}$	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 UNIT	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	Q	R	1, 2, 3
b. RHR Equipment Area Δ Temperature - High	S	Q	R	1, 2, 3
c. Reactor Vessel Water Level - Low Level 3	S	Q	R(b)	1, 2, 3
d. Reactor Vessel Water Level - Low Low Low Level 1	S	Q	R(b)	1, 2, 3
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	S	Q	R(b)	1, 2, 3
f. Drywell Pressure - High	S	Q	R(b)	1, 2, 3
7. MANUAL INITIATION	NA	Q	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 92 days.
(b) Calibrate trip unit setpoint at least once per 92 days.

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.

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 FUNCITONAL 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.

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 required ECCS trip function 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.

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	Q	R(a)	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	Q	R(a)	1, 2, 3
c. LPCS Pump Discharge Flow-Low	S	Q	R(a)	1, 2, 3, 4*, 5*
d. Reactor Vessel Pressure-Low (LPCS/LPCI Injection Valve Permissive)	S	Q	R(a)	1, 2, 3, 4*, 5*
e. LPCI Pump A Start Time Delay Relay	NA	Q	Q	1, 2, 3, 4*, 5*
f. LPCI Pump A Discharge Flow-Low	S	Q	R(a)	1, 2, 3, 4*, 5*
g. LPCS Pump Start Time Delay Relay	NA	Q	Q	1, 2, 3, 4*, 5*
h. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#				
a. Reactor Vessel Water Level - Low Low Low Level 1	S	Q	R(a)	1, 2, 3
b. Drywell Pressure-High	S	Q	R(a)	1, 2, 3
c. ADS Timer	NA	Q	Q	1, 2, 3
d. Reactor Vessel Water Level - Low Level 3	S	Q	R(a)	1, 2, 3
e. LPCS Pump Discharge Pressure-High	S	Q	R(a)	1, 2, 3
f. LPCI Pump A Discharge Pressure-High	S	Q	R(a)	1, 2, 3
g. ADS Drywell Pressure Bypass Timer	NA	Q	Q	1, 2, 3
h. ADS Manual Inhibit Switch	NA	Q	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	Q	R(a)	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	Q	R(a)	1, 2, 3
c. Reactor Vessel Pressure-Low (LPCI Injection Valve Permissive)	S	Q	R(a)	1, 2, 3, 4*, 5*
d. LPCI Pump B Start Time Delay Relay	NA	Q	Q	1, 2, 3, 4*, 5*
e. LPCI Pump Discharge Flow-Low	S	Q	R(a)	1, 2, 3, 4*, 5*
f. LPCI Pump C Start Time Delay Relay	NA	Q	Q	1, 2, 3, 4*, 5*
g. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#				
a. Reactor Vessel Water Level - Low Low Low Level 1	S	Q	R(a)	1, 2, 3
b. Drywell Pressure - High	S	Q	R(a)	1, 2, 3
c. ADS Timer	NA	Q	Q	1, 2, 3
d. Reactor Vessel Water Level - Low Level 3	S	Q	R(a)	1, 2, 3
e. LPCI Pump B and C Discharge Pressure-High	S	Q	R(a)	1, 2, 3
f. ADS Drywell Pressure Bypass Timer	NA	Q	Q	1, 2, 3
g. ADS Manual Inhibit Switch	NA	Q	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	Q	R(a)	1, 2, 3, 4*, 5*
b. Drywell Pressure-High	S	Q	R(a)	1, 2, 3
c. Reactor Vessel Water Level-High Level 8	S	Q	R(a)	1, 2, 3, 4*, 5*
d. Condensate Storage Tank Level - Low	S	Q	R(a)	1, 2, 3, 4*, 5*
e. Suppression Pool Water Level - High	S	Q	R(a)	1, 2, 3, 4*, 5*
f. Pump Discharge Pressure-High	S	Q	R(a)	1, 2, 3, 4*, 5*
g. HPCS System Flow Rate-Low	S	Q	R(a)	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	1, 2, 3, 4**, 5**
b. 4.16kv Standby Bus Under-voltage (Degraded Voltage)	S	M	R	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. 4.16kv 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 92 days.

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.

INSTRUMENTATION

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.

TABLE 3.3.9-1

PLANT SYSTEMS ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]	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

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the TRIP SYSTEM in the tripped condition, provided at least one other OPERABLE channel in the same TRIP SYSTEM is monitoring that parameter.

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	Q	$R^{(a)}$	1, 2, 3
b. Containment-to-Annulus ΔP High	D	Q	$R^{(a)}$	1, 2, 3
c. Reactor Vessel Water Level-Low Low Low Level 1	D	Q	$R^{(a)}$	1, 2, 3
d. Timer	NA	Q	R	1, 2, 3
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM				
a. Reactor Vessel Water Level-High Level 8	D	Q	R	1

(a)Calibrate trip unit setpoint once per 92 days.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm total leakage (averaged over any 24-hour period).

d. 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm leakage at a reactor coolant system pressure of 1025 ± 15 psig from any reactor coolant system pressure isolation valve.

e. 2 gpm UNIDENTIFIED LEAKAGE increase within any period of 24 hours or less (Applicable in OPERATIONAL CONDITION 1 only).

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two other closed manual, deactivated automatic or check* valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the high/low pressure interface valve leakage pressure monitors inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm point at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the following 24 hours. The provisions of Specification 3.0.4 are not applicable.

e. With any reactor coolant system UNIDENTIFIED LEAKAGE increase greater than the limits in e., above, within 4 hours identify the source of leakage as not IGSCC susceptible material or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

* Which have been verified not to exceed the allowable leakage limit at the last refueling outage or after the last time the valve was disturbed, whichever is more recent.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the drywell atmospheric particulate radioactivity at least once per 12 hours,

b. Monitoring the sump flow rates at least once per 12 hours,

c. Monitoring the drywell air coolers condensate flow rate at least once per 12 hours, and

d. Monitoring the reactor vessel head flange leak detection system at least once per 24 hours.

4.4.3.2.2 Each reactor coolant system pressure isolation valve shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 including paragraph IWV-3427(B) of the ASME Code and the RBS Inservice Test Program and verifying the leakage of each valve to be within the specified limit.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

4.4.3.2.3 The high/low pressure interface valve leakage pressure monitors shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

b. CHANNEL CALIBRATION at least once per 18 months.

Tables 3.4.3.2-1 and 3.4.3.2-2 have been deleted.

3/4.4.4 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.4 The chemistry of the reactor coolant system shall be maintained within the limits specified in Table 3.4.4-1.

APPLICABILITY: At all times.

ACTION:

a. In OPERATIONAL CONDITION 1:

1. With the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.4.4-1 for less than 72 hours during one continuous time interval and, for conductivity and chloride concentration, for less than 336 hours per year, but with the conductivity less than 10 μmho/cm at 25°C and with the chloride concentration less than 0.5 ppm, this need not be reported to the Commission and the provisions of Specification 3.0.4 are not applicable.

2. With the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.4.4-1 for more than 72 hours during one continuous time interval or with the conductivity and chloride concentration exceeding the limits specified in Table 3.4.4-1 for more than 336 hours per year, be in at least STARTUP within the next 6 hours.

3. With the conductivity exceeding 10 μmho/cm at 25°C or chloride concentration exceeding 0.5 ppm, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In OPERATIONAL CONDITIONS 2 and 3, with the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.4.4-1 for more than 48 hours during one continuous time interval, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. At all other times:

1. With the:

   a) Conductivity or pH exceeding the limit specified in Table 3.4.4-1, restore the conductivity and pH to within the limit within 72 hours, or

   b) Chloride concentration exceeding the limit specified in Table 3.4.4-1, restore the chloride concentration to within the limit within 24 hours, or

REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 (1) curves A and A' for hydrostatic or leak testing; (2) curves B and B' for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curves C and C' for operations with a critical core other than low power PHYSICS TESTS, with:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 100°F in any one hour period,

c. A maximum temperature change of 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange temperature greater than or equal to 70°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; and determine that the reactor coolant system remains acceptable for continued operations. Otherwise, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined, at least once per 30 minutes, to be within the above required heatup and cooldown limits and to the right of the limit lines of Figure 3.4.6.1-1 curves A and A', B and B', or C and C', as applicable.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curves C and C' within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

4.4.6.1.3 The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties as required by 10 CFR 50, Appendix H. The results of these examinations shall be used to update the curves of Figure 3.4.6.1-1.

4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 70°F:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

   1. ≤ 100°F, at least once per 12 hours.

   2. ≤ 80°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

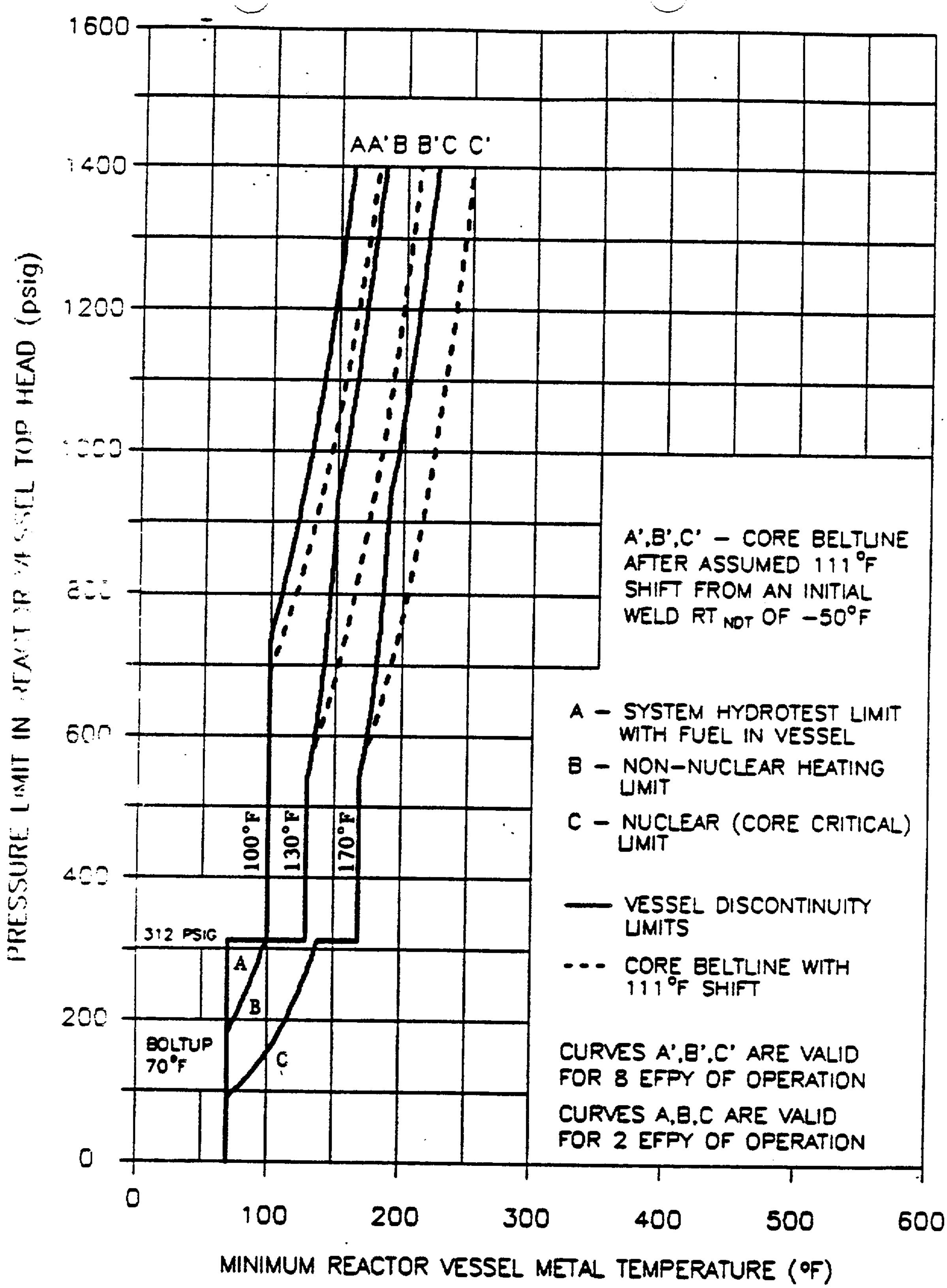

FIGURE 3.4.6.1-1
MINIMUM TEMPERATURE REQUIRED VS REACTOR PRESSURE

Table 4.4.6.1.3-1 has been deleted.

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.3 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to La, 0.26 percent by weight of the primary containment air per 24 hours at Pa, 7.6 psig.

b. A combined leakage rate of less than 0.60 La for all penetrations and all valves subject to Type B and C tests when pressurized to Pa, 7.6 psig.

c. A leakage rate of less than 150 scfh for the valves served by each Division of MS-PLCS and a leakage rate of less than 340 scfh for each of the valve groups identified below when tested in accordance with the surveillance requirements of 4.6.1.3.f.

1. Division I MS-PLCS Valves and Division I PVLCS Valves
2. Division II MS-PLCS Valves and Division II PVLCS Valves
3. Division I MS-PLCS Valves and all first outboard PVLCS Valves

d. A combined leakage rate of less than or equal to 13,500 cc/hr for all penetrations that are annulus bypass leakage paths when pressurized to Pa, 7.6 psig.

e. A combined leakage rate of less than or equal to 170,000 cc/hr, for all valves that are secondary containment bypass leakage paths and equipped with PVLCS, when pressurized to Pa, 7.6 psig.

f. A combined leakage rate of less than or equal to 1 gpm times the total number of containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.1 Pa, 8.36 psig.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY-OPERATING is required per Specification 3.6.1.1.

ACTION:

With:

a. The measured overall integrated primary containment leakage rate equaling or exceeding 0.75 La, or

b. The measured combined leakage rate, for all penetrations and all valves subject to Type B and C tests, exceeding 0.60 La, or

c. The measured leakage rate greater than or equal to 150 scfh for the valves served by each Division of MS-PLCS or the measured leakage rate greater than or equal to 340 scfh for each valve grouping identified in 3.6.1.3.c.1, 3.6.1.3.c.2 or 3.6.1.3.c.3, or

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

d. The combined leakage rate, for all penetrations that are annulus bypass leakage paths, exceeding 13,500 cc/hr, or

e. The combined leakage rate, for all valves that are secondary containment bypass leakage paths and equipped with PVLCS, exceeding 170,000 cc/hr, or

f. The measured combined leakage rate, for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment, exceeding 1 gpm times the total number of such valves,

restore:

a. The overall integrated leakage rate(s) to less than 0.75 La as applicable, and

b. The combined leakage rate, for all penetrations and all valves subject to Type B and C tests, to less than 0.60 La, and

c. The measured leakage rate to less than 150 scfh for the valves served by each Division of MS-PLCS and the measured leakage rate to less than 340 scfh for each of the valve groupings identified in 3.6.1.3.c.1, 3.6.1.3.c.2, and 3.6.1.3.c.3, and

d. The combined leakage rate, for all penetrations that are annulus bypass leakage paths, to less than or equal to 13,500 cc/hr, and

e. The combined leakage rate, for all valves that are secondary containment bypass leakage paths and equipped with PVLCS, to less than or equal to 170,000 cc/hr, and

f. The combined leakage rate, for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment, to less than or equal to 1 gpm times the total number of such valves,

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.3 The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4 (1972):

a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 * 10 month intervals during shutdown at Pa, 7.6 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet 0.75 La, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 La, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 La, at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

   1. Confirms the accuracy of the test by verifying that the difference between the supplemental test data and the Type A test data is within 0.25 La. The formula to be used is: $[Lo + Lam - 0.25\ La] \leq Lc \leq [Lo + Lam + 0.25\ La]$ where Lc = supplemental test results; Lo = superimposed leakage; Lam = measured Type A leakage.

   2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

   3. Requires the quantity of gas, injected into the primary containment or bled from the primary containment during the supplemental test, to be between 0.75 La and 1.25 La.

d. Type B and C tests shall be conducted with gas at Pa, 7.6 psig*, at intervals no greater than 24 months except for tests involving:

   1. Air locks,
   2. Main steam positive leakage control system (MS-PLCS) valves and PVLCS valves,
   3. Penetrations using continuous leakage monitoring systems,
   4. Primary containment isolation valves in hydrostatically tested lines which penetrate the primary containment, and
   5. Purge supply and exhaust isolation valves with resilient material seals.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.4.

f. Total sealing air leakage into the primary containment, at a test pressure of 11.5 psid for MS-PLCS valves and 33 psid for penetration leakage control system sealed valves, shall be determined by test at least once per 18 months. This leakage may be excluded when determining the combined leakage rate, 0.6 La.

*Unless a hydrostatic test is required.

g. Type B tests for electrical penetrations employing a continuous leakage monitoring system shall be conducted at Pa, 7.6 psig, at intervals no greater than once per 3 years.

h. Leakage from isolation valves that are sealed with the PVLCS shall be tested once per 24 months with the valves pressurized to at least Pa, 7.6 psig. This leakage may be excluded when determining the combined leakage rate, 0.6 La.

i. Primary containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at least once per 18 months. This leakage may be excluded when determining the combined leakage rate, 0.6 La.

j. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.9.3.

k. The provisions of Specification 4.0.2 are not applicable to Specifications 4.6.1.3.a, 4.6.1.3.d, 4.6.1.3.g, and 4.6.1.3.h.

Table 3.6.1.3-1 has been deleted.

LIMITING CONDITION FOR OPERATION

3.6.1.4 Each primary containment air lock shall be OPERABLE with:

a. Both doors closed, except at least one air lock door shall be closed when the air lock is being used for normal transit entry and exit through the containment, and

b. An overall air lock leakage rate in compliance with the limits of Specification 3.6.1.3.d when pressurized to Pa, 7.6 psig, and

c. The inflatable seal system air flask pressure $\geq$ 90 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2*, 3, and #.

ACTION:

a. With one primary containment air lock door in one or both air locks inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed except as provided in a.3.

2. With an air lock door in only one air lock inoperable, operation may then continue until performance of the next required overall air lock leakage test** provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. With one air lock door in each air lock inoperable, operation may then continue with entry and exit permitted for up to 7 days** provided that an OPERABLE air lock door is verified to be locked closed after each entry or exit and an individual is dedicated to assure that two doors in an air lock are not opened simultaneously.

4. Otherwise, in OPERATIONAL CONDITIONS 1, 2, or 3 be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

5. Otherwise, in Operational Condition #, suspend all operations involving handling of irradiated fuel in the containment, CORE ALTERATIONS, and operations with a potential for draining the reactor vessel.

*See Special Test Exception 3.10.1.

**The provisions of Specification 3.0.4 are not applicable.

#When irradiated fuel is being handled in the primary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

3/4.6.4 PRIMARY CONTAINMENT AND DRYWELL ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 Each primary containment and drywell isolation valve shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With one or more of the primary containment or drywell isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and, within 4 hours, either:

a. Restore the inoperable valve(s) to OPERABLE status, or

b. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position, and declare the associated system inoperable, if applicable, and perform the associated ACTION statements for that system,* or

c. Isolate each affected penetration by use of at least one closed manual valve or blind flange and declare the associated system inoperable, if applicable, and perform the associated ACTION statements for that system.*

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each primary containment or drywell isolation valve shall be demonstrated OPERABLE prior to returning the valve to service, after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit, by cycling the valve through at least one complete cycle of full travel and verifying the isolation time.

4.6.4.2 Each automatic primary containment or drywell isolation valve shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that, on an isolation test signal, each automatic isolation valve actuates to its isolation position.

*Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative controls.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.4.3 The isolation time of each power operated or automatic valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

Table 3.6.4-1 has been deleted.

(Next page is 3/4 6-48)

CONTAINMENT SYSTEMS

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.

CONTAINMENT SYSTEMS

SECONDARY CONTAINMENT AUTOMATIC ISOLATION DAMPERS

LIMITING CONDITION FOR OPERATION

3.6.5.3 Each secondary containment ventilation system automatic isolation damper shall be OPERABLE.

APPLICABILITY: - OPERATIONAL CONDITIONS 1, 2, 3.
- FOR FUEL BUILDING DAMPERS: OPERATIONAL CONDITIONS 1, 2, 3, AND ##.

ACTION:

With one or more of the secondary containment ventilation system automatic isolation dampers inoperable, maintain at least one isolation damper OPERABLE in each affected penetration that is open and, within 8 hours, either:

a. Restore the inoperable damper(s) to OPERABLE status, or

b. Isolate each affected penetration by use of at least one deactivated automatic damper secured in the isolation position and declare the associated system inoperable, if applicable, and perform the associated ACTION statements for that system, or

c. Isolate each affected penetration by use of at least one closed manual valve or blind flange and declare the associated system inoperable, if applicable, and perform the associated ACTION statements for that system.

Otherwise, in OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Otherwise, in Operational Condition ##, suspend handling of irradiated fuel in the Fuel Building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.3 Each secondary containment ventilation system automatic isolation damper shall be demonstrated OPERABLE:

a. Prior to returning the damper to service after maintenance, repair or replacement work is performed on the damper or its associated actuator, control or power circuit, by cycling the damper through at least one complete cycle of full travel and verifying the isolation time.

##When irradiated fuel is being handled in the Fuel Building.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months, during COLD SHUTDOWN or REFUELING, by verifying that, on a secondary containment isolation test signal, each secondary containment automatic isolation damper actuates to its isolation position.

c. By verifying the isolation time to be within its limit when tested pursuant to Specification 4.0.5.

Table 3.6.5.3-1 has been deleted.

ELECTRICAL POWER SYSTEMS

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 Each of the primary and backup overcurrent protective devices associated with each primary containment electrical penetration circuit shall be OPERABLE. The scope of these protective devices excludes those circuits for which credible fault currents would not exceed the electrical penetrations' design ratings.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With one or more of the overcurrent protective devices inoperable, declare the affected system or component inoperable and apply the appropriate ACTION statement for the affected system and:

a. For 4.16 kV circuit breakers, de-energize the 4.16 kV circuit(s) by tripping the associated redundant circuit breaker(s) within 72 hours and verifying, at least once per 7 days thereafter, the redundant circuit breaker to be tripped.

b. For 480 volt circuit breakers, remove the inoperable circuit breaker(s) from service by racking out the breaker within 72 hours and verifying, at least once per 7 days thereafter, the inoperable breaker(s) to be racked out.

c. For 480 volt MCC circuit breaker/fuse combination starters, remove the inoperable starter(s) from service by locking the breakers open and removing the control power fuse within 72 hours and verifying, at least once per 7 days thereafter, the inoperable starter(s) circuit breaker to be locked open with the control power fuse removed.

d. For 120/140 volt molded case circuit breakers, remove the inoperable circuit breaker(s) from service by tripping both 120/140 volt breakers open and locking the upstream 480 volt MCC breaker open within 72 hours and verifying, at least once per 7 days thereafter, the 480 volt MCC breaker(s) to be locked open.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.8.4.1 Each of the overcurrent protective devices shall be demonstrated OPERABLE:

a. At least once per 18 months:

1. By verifying that the medium voltage 4.16 kv circuit breakers are OPERABLE by selecting, on a rotating basis, at least one of the four circuit breakers and performing:

a) A CHANNEL CALIBRATION of the associated protective relays, and

b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and overcurrent control circuits function as designed.

c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least one of the four circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting currents in excess of the breaker's nominal setpoint and measuring the response time of the long time and short time delay elements and the setpoint of the instantaneous element, as appropriate. The measured data shall be compared to the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

3. By selecting and functionally testing a representative sample of at least 10% of each type of motor starter used for penetration redundant overcurrent protection. Motor starters selected for functional testing shall be selected on a rotating basis. Testing of these motor starters shall consist of injecting a current with a value equal to the locked rotor current of the associated motor and verifying that the motor starter operates to interrupt the current within the associated thermal overload time delay band width for that current as specified by the manufacturer. Motor starters found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each motor starter found inoperable during these functional tests, an additional representative sample of at least 10% of all the motor starters of the inoperable type shall also be functionally tested until no more failures are found or all motor starters of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance program in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

Table 3.8.4.1-1 has been deleted.

OTHER OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.2 Each primary overcurrent protection device for the Main Control Room safety-related lighting and the primary and secondary RPS Alternate Source of Power shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

With one or more of the overcurrent protective devices inoperable, remove the circuit breaker(s) feeding the control room lighting and/or alternate RPS supply as appropriate from service by opening the breaker(s) within 72 hours and return the overcurrent protection devices to OPERABLE status within 7 days, or verify the appropriate breakers open at least once per 24 hours.*

SURVEILLANCE REQUIREMENTS

4.8.4.2 The overcurrent protective devices shall be demonstrated OPERABLE at least once per 18 months by selecting and testing one-half of each type of circuit breaker on a rotating basis. Testing of these circuit breakers shall consist of injecting currents in excess of the breaker's nominal setpoint and measuring the response time of the long time and short time delay elements and the setpoint of the instantaneous element, as appropriate. The measured data shall be compared to the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer.

*Except at least once per 31 days if locked, sealed, or otherwise secured in the open position.

Table 3.8.4.2-1 has been deleted.

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.

A.C. CIRCUITS INSIDE CONTAINMENT

LIMITING CONDITION FOR OPERATION

3.8.4.4 Each 480 V and 240/120 V A.C. circuit inside containment for the Containment Building HVAC, Drywell Cooling HVAC, RWCU, Inclined Fuel Transfer Tube, and Reactor Building Main Hoist systems without redundant penetration protection shall be de-energized*.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With any A.C. circuit energized, trip the associated circuit breaker(s) in the specified location within 1 hour.

SURVEILLANCE REQUIREMENTS

4.8.4.4 Each A.C. circuit 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.

RESPONSIBILITIES (Continued)

c. Provide written notification within 24 hours to the Senior Vice President - RBNG and the Nuclear Review Board of disagreement between the FRC and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.1.8 The FRC shall maintain written minutes of each FRC meeting that, at a minimum, document the results of all FRC activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the Plant Manager and the NRB.

6.5.2 TECHNICAL REVIEW AND CONTROL

6.5.2.1 Each procedure and program required by Specification 6.8 and other procedures that affect nuclear safety, and changes thereto, is prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group that prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group that prepared the procedure. Each such procedure and program, or changes thereto, shall be approved, prior to implementation, by the Plant Manager, one of the Assistant Plant Managers or the Director - Radiological Programs, or the manager/department head responsible for the program or the activity described in the procedure, with the exception of the Emergency Plan and implementing procedures which shall be approved by the Manager - Administration, Plant Manager and Senior Vice President - RBNG.

6.5.2.2 Individuals responsible for reviews performed in accordance with Section 6.5.2.1 shall be members of River Bend Nuclear Group supervisory staff, and the reviews shall be performed in accordance with administrative procedures. Each such review shall include a determination of whether or not additional, cross-disciplinary review is necessary and a verification that the proposed actions do not constitute an unreviewed safety question. If deemed necessary, such review shall be performed by the appropriate designated review personnel.

6.5.2.3 The station security program and implementing procedures shall be reviewed at least once per 12 months, and recommended changes approved in accordance with Specification 6.5.2.1.

6.5.2.4 The station emergency plan and implementing procedures and recommended changes shall be approved in accordance with Specification 6.5.2.1.

6.5.2.5 The station fire protection plan and implementing procedures shall be reviewed at least once per 12 months, and recommended changes approved in accordance with Specification 6.5.2.1.

6.5.2.6 The station Technical Requirements Manual and implementing procedures and recommended changes shall be approved in accordance with Specification 6.5.2.1.

6.5.2.7 Records documenting each of the activities performed under Specifications 6.5.2.1 through 6.5.2.6 shall be maintained.

6.5.3 NUCLEAR REVIEW BOARD (NRB)

FUNCTION

6.5.3.1 The NRB shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,
b. Nuclear engineering,
c. Chemistry and radiochecmistry,
d. Metallurgy,
e. Instrumentation and control,
f. Radiological safety,
g. Mechanical and electrical engineering,
h. Quality assurance practices,
i. Licensing and regulatory affairs,
j. Training.

The NRB shall report to and advise the Senior Vice President - RBNG on those areas of responsibility in Specifications 6.5.3.7 and 6.5.3.8.

COMPOSITION

6.5.3.2 The NRB shall be composed of at least nine but not more than thirteen individuals who shall possess the necessary expertise to provide the independent review and audit functions identified in Specification 6.5.3.1. All members shall be qualified to the applicable portions of ANSI/ANS 3.1-1978, Section 4.7, prior to being approved by the Chairperson. This qualification will be maintained while assigned to NRB activities. The Senior Vice President-RBNG provides nominations for permanent NRB membership to the NRB Chairperson for review and approval.

ALTERNATES

6.5.3.3 All alternate members shall be appointed in writing by the NRB Chairperson and be qualified to the applicable portions of ANSI/ANS 3.1-1978, Section 4.7, prior to being approved by the Chairperson. This qualification will be maintained while assigned to NRB activities.

CONSULTANTS

6.5.3.4 Consultants shall be utilized as determined by the NRB Chairperson to provide expert advice to the NRB.

MEETING FREQUENCY

6.5.3.5 The NRB shall meet at least once per 6 months.

RECORDS (Continued)

a. Minutes of each NRB meeting shall be prepared, approved, and forwarded to the Senior Vice President - RBNG within 14 days following each meeting.

b. Reports of reviews encompassed by Specification 6.5.3.7 shall be prepared, approved, and forwarded to the Senior Vice President - RBNG within 14 days following completion of the review.

c. Audit reports encompassed by Specification 6.5.3.8 shall be forwarded to the Senior Vice President - RBNG, and to the management positions responsible for the areas audited, within 30 days after completion of the audit by the auditing organization.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of 10 CFR 50.73 and

b. Each REPORTABLE EVENT shall be reviewed by the FRC and the results of this review shall be submitted to the NRB and the Plant Manager.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Senior Vice President - RBNG and the NRB chairman (or personnel acting for their function) shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the FRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the NRB, and the Senior Vice President - RBNG within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

PROCEDURES AND PROGRAMS (Continued)

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978.

b. The applicable procedures required to implement the requirements of NUREG-0737 and supplements thereto.

c. Refueling operations.

d. Surveillance and test activities of safety-related equipment.

e. Security Plan implementation.

f. Emergency Plan implementation.

g. Fire Protection Program implementation.

h. Process Control Program implementation.

i. Offsite Dose Calculation Manual implementation.

j. Quality Assurance Program for effluent and environmental monitoring.

k. Technical Requirements Manual implementation.

6.8.2 Each procedure of Specification 6.8.1, and changes thereto, shall be reviewed and approved in accordance with Specification 6.5.2.1.

6.8.3 Temporary changes to procedures of Specification 6.8.1 may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Operator license on the unit affected; and

c. The change is documented, reviewed by the FRC as required by Specification 6.5.1.6, and approved in accordance with Specification 6.5.2.1 within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage, from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident, to as low as practicable levels. The systems include the HPCS, LPCS, RHR, RCIC, process sampling, and standby gas treatment systems. The program shall include the following:

1. Preventive maintenance and periodic visual inspection requirements, and

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT 76 TO FACILITY OPERATING LICENSEE NO. NPF-47

ENTERGY OPERATIONS, INC.

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

1.0 INTRODUCTION

By letter dated January 14, 1994, as supplemented by letters dated November 10, 1994, and February 8, 1995, Entergy Operations, Inc. (EOI), (the licensee), requested an amendment to Facility Operating License No. NPF-47 for the River Bend Station (RBS). The proposed amendment would remove the technical specification (TS) tables that include lists of components referenced in individual specifications and relocates the reactor vessel material specimen withdrawal schedule from the TSs to the updated safety analysis report (USAR). Additionally, the TSs have been modified such that all references to these tables have been removed. Finally, the TSs have been modified to state requirements in general terms that include the components listed in the tables being removed from the TSs. Guidance on the proposed TS changes was provided by Generic Letter (GL) 91-08, "Removal of Component Lists from Technical Specifications," dated May 6, 1991.

The February 8, 1995, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination

2.0 DISCUSSION

Section 50.36 of Title 10 of the Code of Federal Regulations established the regulatory requirements related to the content of TSs. The rule requires that TSs include items in specific categories, including safety limits, limiting conditions for operation, and surveillance requirements; however the rule does not specify the particular requirements to be included in a plant's TSs. The NRC developed criteria, as described in the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Plants" (58 FR 39132), to determine which of the design conditions and associated surveillances need to be located in the TSs. The final policy statement adopted the subjective statement of the Atomic Safety and Licensing Appeal Board, ALAB 531, 9 NRC 263, 273 (1979), Portland General Electric Co. (Trojan Nuclear Plant) as the basis for the criteria. The Appeal Board stated,

> "... there is neither a statutory nor a regulatory requirement that every operational detail set forth in an applicant's safety analysis report (or equivalent) be subject to a technical specification, to be included in the license as an absolute condition of operation which is legally binding upon the licensee unless and until changed with specific Commission approval. Rather, as best we can discern it, the contemplation of both the Act and the regulations is that technical specifications are to be reserved for those matters as to which the

imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety."

Briefly, the criteria provided by the final policy statement are (1) detection of abnormal degradation of the reactor coolant pressure boundary, (2) boundary conditions for design basis accidents and transients, (3) primary success paths to prevent or mitigate design basis accidents and transients, and (4) functions determined to be important to risk or operating experience. The Commission's final policy statement acknowledged that its implementation may result in the relocation of existing TS requirements to licensee controlled documents and programs.

GL 91-08 provides specific guidance on the removal of component lists from TSs when removal of the component lists do not alter existing TS requirements or those components to which they apply. GL 91-01 provides guidance on relocating the reactor vessel specimen withdrawal schedule from the TSs to the USAR.

3.0 EVALUATION

In accordance with GL 91-01, GL 91-08, and 10 CFR 50.90, the licensee proposed the following changes to the RBS TS. The licensee's proposed changes are discussed in the order in which the associated specification appears in the RBS Technical Specifications. The staff's evaluation and conclusion follow each proposed change.

(1) The TS index pages are being revised to make editorial corrections to reflect the deletion of tables which contain component lists or to reflect renumbering of pages due to deletion of large lists. It is proposed that the deleted pages be retained as denoted in the marked-up pages.

The staff concludes that the proposed changes are acceptable based on the fact that they are administrative in nature only (reflecting the TS changes evaluated below).

(2) It is proposed that Definition 1.32, PRIMARY CONTAINMENT INTEGRITY - FUEL HANDLING, Items (a), (b), and (c), and Definition 1.33, PRIMARY CONTAINMENT INTEGRITY - OPERATING, Items (a), (b), (c), and (d), be revised by adding "primary" to each of the appropriate subparagraphs for clarification.

These changes are provided for clarification only and do not result in any technical change to the current TSs. Therefore the staff finds the proposed changes acceptable.

(3) The licensee proposes to delete Footnote (***) to Surveillance Requirement (SR) 4.3.1.2 and Footnote (p) to Table 4.3.1.1-1, associated with Reactor Vessel Steam Dome Pressure - High and Main Steam Line Radiation - High (Functional Units 3 and 7).

Both footnotes refer to extensions associated with the first refueling outage, which occurred in 1987, and are no longer necessary. This proposed change is editorial and does not result in a technical change to the current requirements. Therefore, the staff finds the proposed changes acceptable.

(4) The licensee proposes to delete Footnote (*) to SR 4.3.2.2 and Footnotes (c) and (d) to Table 4.3.2.1-1, associated with Reactor Vessel Water Level - Low Low Level 2, Reactor Vessel Water Level - Low Low Low Level 1, Main Steam Line Radiation - High, Main Steam Line Flow - High, Reactor Vessel (RHR [residual heat removal] Cut-in Permissive) Pressure - High, and Drywell Pressure - High (Trip Functions 1.a, 2.a, 2.b, 2.d, 3.a, 4.e, 6.d, 6.e, and 6.f).

All three footnotes refer to extensions associated with the first refueling outage, which occurred in 1987, and are no longer necessary. This proposed change is editorial and does not result in a technical change to the current requirements. Therefore, the staff finds the proposed changes acceptable.

(5) The licensee proposes to delete the "Valve Groups Operated By Signal" column and Footnote (***) from Table 3.3.2-1; revise Footnote (e) to Table 3.3.2-1 to delete reference to Table 3.6.5.3-1; and relocate Footnotes (a) through (l) from the column "Valve Groups Operated By Signal" portion of Table 3.3.2-1 to the associated individual trip function.

The "VALVE GROUPS OPERATED BY SIGNAL" column, through reference to Tables 3.6.4-1 and 3.6.5.3-1, identifies which valve group(s) each isolation trip function affects and includes a number of notes which are associated with the trip function or the valve group(s). The valve groups are proposed to be deleted from the TSs and relocated to the Technical Requirements Manual (TRM). The valve groups identified in this TS are for information only and require a listing of the individual valves (associated with the groups) to be of use. The current listing of individual valves (associated with the groups) is currently provided in Table 3.6.4-1, "Containment and Drywell Isolation Valves" and Table 3.6.5.3-1, "Secondary Containment Ventilation System Automatic Isolation." These tables are also being deleted from the TSs and relocated to the TRM (see below). The valve groups are not necessary to support the actions or surveillances required by the TSs.

Footnote (***) to Table 3.3.2-1 is proposed to be deleted from the TSs and relocated to the TRM. This note refers the reader to Tables 3.6.4-1 and 3.6.5.3-1 for the list of affected valves. A listing of the valves in the TSs is not necessary to support the actions or surveillances required by the TS.

Footnote (e) to Table 3.3.2-1 is proposed to be revised to delete the reference to Table 3.6.5.3-1, which lists the secondary containment ventilation system automatic isolation dampers. Table 3.6.5.3-1 is being relocated to the TRM. With the proposed revision, the note identifies in general terms that isolation dampers are also actuated by the trip function.

The proposed changes are administrative only, reflecting the deletion of Tables 3.6.4-1 and 3.6.5.3-1. Therefore the staff finds the proposed changes acceptable.

(6) The licensee proposes to revise Footnote (#) to Table 3.3.2-3 to delete reference to Tables 3.6.4-1 and 3.6.5.3-1.

Tables 3.6.4-1 and 3.6.5.3-1 are being relocated to the TRM (as discussed below). With the proposed revision to the footnote, the SR has been stated in general terms to apply to all valve isolation times. Additional guidance will now be found in the tables relocated to the TRM. Therefore, the staff finds the proposed changes acceptable.

(7) The licensee proposes to delete Footnote (##) to SR 4.3.3.2 and SR 4.3.3.3, and Footnote (b) to Table 4.3.3.1-1, associated with Reactor Vessel Water Level - Low Low Low Level 1, Drywell Pressure - High, LPCS [Low-Pressure Core Spray] Pump Discharge Flow - Low, LPCI [Low-Pressure Coolant Injection] Pump Discharge Flow - Low, Reactor Vessel Pressure - Low, LPCI Pump Start Time Delay Relay, LPCS Pump Start Time Delay Relay, LPCI/LPCS Manual Initiation, and 4.16 kV Standby Bus Undervoltage (Trip Functions A.1.a, A.1.b, A.1.c, A.1.d, A.1.e, A.1.f, A.1.g, A.1.h, B.1.a, B.1.b, B.1.c, B.1.d, B.1.e, B.1.f, B.1.g, D.1.a, and D.1.b).

Footnote (##) referred to a "Note C" in Table 4.3.3.1-1 that had concerned the first refueling outage and had previously been deleted. Footnote B referred to an extension for the fifth refueling outage, which ended on June 29, 1994. These footnotes are no longer necessary and can be removed since the subject refueling outages have been completed. This proposed change is editorial in nature and does not result in a technical change to the current requirements. The staff finds the proposed changes acceptable.

(8) The licensee proposes to delete Footnote (#) to SR 4.3.9.2 and Table 4.3.9.1-1, associated with Reactor Vessel Water Level - Low Low Low Level and Drywell Pressure - High (Trip Functions 1.c and 1.a).

These footnotes permitted an extension of the 18 month test interval until the completion of the first refueling outage, which ended in 1987, and are no longer necessary. This proposed change is editorial in nature and does not result in a technical change to the current requirements. The staff finds the proposed changes acceptable.

(9) The licensee proposes to revise Limiting Condition for Operation (LCO) 3.4.3.2.d and SR 4.4.3.2.2 to delete references to Table 3.4.3.2-1.

The proposed revisions delete the references to Table 3.4.3.2-1, "Reactor Coolant System Pressure Isolation Valves," which identifies those valves. The referenced Table 3.4.3.2-1 is being deleted from the TS and relocated to the TRM (as discussed below). With the proposed revision, the LCO and the SR state that the requirements in each apply to reactor coolant pressure isolation valves, in general. This proposed change is administrative in nature reflecting the relocation of Table 3.4.3.2-1 to the TRM. The staff finds the proposed change acceptable.

(10) The licensee proposes to revise ACTION Statement 3.4.3.2.d to delete reference to Table 3.4.3.2-2 and SR 4.4.3.2.3 to delete reference to alarm setpoints contained in Table 3.4.3.2-2.

ACTION Statement 3.4.3.2.d is proposed to be revised to delete the reference to Table 3.4.3.2-2, "Reactor Coolant System Interface Valves Leakage Pressure Monitors," which is a list of those monitors. SR 4.4.3.2.3 is also proposed to be revised to delete the reference to alarm setpoints contained in Table 3.4.3.2-2. Table 3.4.3.2-2, including alarm setpoints, is being deleted from the TS and relocated to the TRM. The setpoints are not Limited Safety Settings as defined by 10 CFR 50.36 and the proposed control by the licensee is acceptable. With the proposed revision, the action statement and the SR state the requirements in each apply to interface valve leakage pressure monitors in general terms. This proposed change is administrative in nature, reflecting the relocation of Table 3.4.3.2-2 to the TRM. The staff finds the proposed changes acceptable.

(11) Table 3.4.3.2-1, "Reactor Coolant System Pressure Isolation Valves," is proposed to be deleted from the TS and relocated to the TRM. Table 3.4.3.2-1 currently denotes the valve number, function, and associated system for the reactor coolant system (RCS) pressure isolation valves.

In July 1993, the NRC issued NUREG-1463, "Regulatory Analysis for the Resolution of Generic Safety Issue 105: Interfacing System Loss-of-Coolant Accident (LOCA) in Light-Water Reactors." This report concluded that the interfacing system LOCA, which is the rationale for the RCS pressure isolation valves being in plant system design, is not a risk concern for boiling water reactors (BWRs). This is also supported by the RBS Individual Plant Examination (IPE) for GL 88-20 which closes plant specific concerns relating to Generic Safety Issue 105. The results of the RBS IPE conclude that the chance of an interfacing system LOCA causing a harsh operating environment for emergency core cooling system (ECCS) equipment was negligibly small and the risk posed by an interfacing system LOCA, given normal operating environment for ECCS equipment (and thus generic failure probabilities for ECCS), is very small.

The staff's review of the proposed change determined that the relocation of Table 3.4.3.2-1 does not eliminate the requirements for the licensee to ensure that the RCS pressure isolation valves are capable of performing their safety function. Although Table 3.4.3.2-1 is relocated from the TSs to the TRM, the information being relocated will be controlled and subsequent changes reviewed in accordance with the change control program described in TS 6.5.2. The staff finds the proposed change acceptable.

(12) Table 3.4.3.2-2 is also proposed to be deleted from the TSs and relocated to the TRM. Table 3.4.3.2-2 currently denotes instrument numbers, functions, and alarm setpoints for RCS high/low pressure interface valve leakage pressure monitors. Alarm setpoints are typically addressed in plant operational procedures, however, these setpoints will be relocated to the TRM. These alarm setpoints are not Limited Safety Settings as defined by 10 CFR 50.36 and the proposed control by the licensee is acceptable.

The information being relocated to the TRM from this table is controlled and subsequent changes reviewed in accordance with the change control program described in Specification 6.5.2. As a result, the deletion and subsequent relocation of this information from the TSs complies with the guidance contained in GL 91-08. The proposed change, therefore, does not result in a technical change to the TSs. The staff finds the proposed change acceptable.

(13) SR 4.4.6.1.3 is proposed to be revised to delete the reference to Table 4.4.6.1.3-1, "Reactor Vessel Material Surveillance Program - Withdrawal Schedule."

The referenced Table 4.4.6.1.3-1 is being deleted from the TSs and relocated to the TRM (as discussed below). The reference to the table is for information only and is not necessary to support the surveillance required by the TSs. The staff finds the proposed change acceptable.

(14) Table 4.4.6.1.3-1, "Reactor Vessel Material Surveillance Program - Withdrawal Schedule," is proposed to be deleted from the TSs and relocated to the TRM.

As stated in GL 91-01, this schedule is redundant to 10 CFR Part 50, Appendix H (which requires prior NRC approval of changes to the specimen withdrawal schedule). Therefore, it is unnecessary to retain this table in the TSs. The specimen withdrawal schedule is not necessary to support the actions or surveillances required by the TSs. Since the table would be relocated to the TRM, appropriate notations will be added to the TRM to clarify that prior NRC approval of changes to the specimen withdrawal schedule is required. Additionally, the capsule numbers, vessel locations, lead factors, and withdrawal schedule are described in Updated Safety Analysis Report (USAR) Section 5.3.1.6. The deletion and subsequent relocation of this information from the TSs complies with the guidance contained in GL 91-01. The staff finds the proposed change acceptable.

(15) The licensee proposes to revise LCO 3.6.1.3.b to delete the reference to Table 3.6.4-1 for appropriate test pressures for Type B and Type C tests. With the removal of this table which required 7.6 psig, the licensee proposes to modify the LCO to add a test pressure value of 7.6 psig. This proposed change to add the pressure in the LCO will maintain the requirements suitable for the facility in the TSs and will allow lists to be relocated.

This test pressure is currently stated in the SRs and therefore there is no technical change to the LCO. The staff finds the proposed changes acceptable.

(16) LCO 3.6.1.3.d, ACTION Statement (with) 3.6.1.3.d, and ACTION Statement (restore) 3.6.1.3.d are proposed to be revised to delete the references to Table 3.6.1.3-1 "Annulus Bypass Leakage Paths". Table 3.6.1.3-1 is being deleted from the TSs and relocated to the TRM (as discussed below).

This is an administrative change to support the deletion of Table 3.6.1.3-1. With the proposed revisions, the requirements apply to all primary containment penetrations that are annulus bypass leakage paths, rather than those specified in the table. The staff finds the proposed changes acceptable.

(17) LCO 3.6.1.3.e, ACTION (with) Statement 3.6.1.3.e, and ACTION (restore) Statement 3.6.1.3.e are proposed to be revised to delete the references to Table 3.6.4-1, "Containment and Drywell Isolation Valves". The referenced Table 3.6.4-1 is being deleted from the TSs and relocated to the TRM (as discussed below).

This is an administrative change reflecting the deletion of Table 3.6.4-1. With the proposed revisions, the requirements apply to all valves that are secondary containment bypass leakage paths and are equipped with penetration valve leakage control system (PVLCS), rather than those valves specified in the table. The staff finds the proposed changes acceptable.

(18) LCO 3.6.1.3.f, ACTION (with) Statement 3.6.1.3.f, ACTION (restore) Statement 3.6.1.3.f, SR 4.6.1.3.d.4, SR 4.6.1.3.i, and Footnote (*) to SR 4.6.1.3.d are proposed to be revised to delete the references to Table 3.6.4-1, "Containment and Drywell Isolation Valves". The referenced Table 3.6.4-1 is being deleted from the TSs and relocated to the TRM (as discussed below).

This is an administrative change reflecting the deletion of Table 3.6.4-1. With the proposed revisions, the requirements apply to all primary containment isolation valves in hydrostatically tested lines which penetrate containment, rather than those specified in the table. The staff finds the proposed changes acceptable.

(19) Delete Footnote (**) to SR 4.6.1.3.d and SR 4.6.1.3.f.

This footnote provides an extension of the test interval until the first refueling outage, which occurred in 1987, and is no longer necessary. The proposed change is editorial in nature and does not result in a technical change to the current requirements. The staff finds the proposed change acceptable.

(20) Delete Footnote (*) to SR 4.6.1.3.i.

This footnote provides an extension of the test interval until the first refueling outage, which occurred in 1987, and is no longer necessary. The proposed change is editorial in nature and does not result in a technical change to the current requirements. The staff finds the proposed change acceptable.

(21) Table 3.6.1.3-1 is proposed to be deleted from the TSs and relocated to the TRM. Table 3.6.1.3-1 currently denotes annulus bypass leakage paths associated with Specification 3/4.6.1.3 "Primary Containment Leakage". The table also specifies leakage limits for the fuel building and auxiliary building. The combined leakage limit is specified in LCO 3.6.1.3.d as 13,500 cc/hr for both the fuel building and auxiliary building.

With the proposed revisions to TS 3.6.1.3 (see above), the requirements for annulus bypass leakage paths are stated in general terms and the details in Table 3.6.1.3-1 are not necessary to support the actions or surveillances required by the TSs. The information being relocated to the TRM is controlled and subsequent changes will be reviewed in accordance with the change control program described in Specification 6.5.2. The relocation of the information in the table complies with the guidance contained in GL 91-08. The staff finds the proposed change acceptable.

(22) SR 4.6.1.3.i is proposed to be revised to include part of note (j) from Table 3.6.4-1. This addition is to clarify the existing SR since the reference to Table 3.6.4-1 was removed (as discussed above). Although, this statement is a duplication of the 10 CFR Part 50, Appendix J requirements, it is proposed to be added to SR 4.6.1.3.i as follows for clarity.

> "This leakage may be excluded when determining the combined leakage rate, 0.60 La."

This is an administrative change to support deletion of Table 3.6.4-1. The staff finds the proposed changes acceptable.

(23) LCO 3.6.4 is proposed to be revised to delete the reference to Table 3.6.4-1 "Containment and Drywell Isolation Valves" and the reference to the isolation times shown in Table 3.6.4-1. In addition, the proposed revision makes the associated requirements applicable to each containment and drywell isolation valve, rather than only those specified in the table. The table is being deleted from the TS and relocated to the TRM (as discussed below).

With the proposed revisions, the requirements apply to containment and drywell isolation valves in general and the information from Table 3.6.4-1 is not necessary to support the actions or SR by the TSs. The information being relocated to the TRM is controlled and subsequent changes will be reviewed in accordance with the change control program described in Specification 6.5.2. The relocation of the information in the table complies with the guidance contained in GL 91-08. The staff finds the proposed change acceptable.

(24) ACTION Statement 3.6.4, SR 4.6.4.1, SR 4.6.4.2, and SR 4.6.4.3 are proposed to be revised to delete the references to Table 3.6.4-1 "Containment and Drywell Isolation Valves". The referenced Table 3.6.4-1 is being deleted from the TSs and relocated to the TRM (as discussed below).

This is an administrative change to support the deletion of Table 3.6.4-1. With the proposed revision, the requirements apply to primary containment and drywell isolation valves in general, rather than only to those valves specified in the table. The staff finds the proposed changes acceptable.

(25) SR 4.6.4.1 is proposed to be revised to delete the word "specified" since, with the deletion of Table 3.6.4-1 (Containment and Drywell Isolation Valves), the isolation times for automatic primary containment isolation valves (PCIVs) are no longer specified in the TS. Table 3.6.4-1 is being deleted from the TSs and relocated to the TRM (as discussed below).

This is an editorial change to support the deletion of Table 3.6.4-1 and the revision discussed above. The staff finds the proposed changes acceptable.

(26) Table 3.6.4-1 is proposed to be deleted from the TSs and relocated to the TRM. Table 3.6.4-1 currently denotes drywell and containment isolation valves associated with Specification 3/4.3.2 (Isolation Actuation Instrumentation), Specification 3/4.6.1.3 (Primary Containment Leakage) and Specification 3/4.6.4 (Primary Containment and Drywell Isolation Valves). The table also includes a list of associated valve numbers, penetration numbers, valve groups, maximum isolation times, and containment penetrations which constitute secondary containment bypass leakage paths. The footnotes associated with Table 3.6.4-1 are also proposed to be relocated to the TRM. Footnote (j) is the only footnote to Table 3.6.4-1 being retained, in part (as discussed above).

With the proposed changes to TSs 3.3.2, 3.6.1, and 3.6.4 that generalize the requirements in each TS, the information in the table is not necessary to support the actions or the surveillances required by the TSs. The information contained in the table is also described in USAR Table 6.2-40 and will be in the TRM. The information being relocated to the TRM will be controlled and subsequent changes will be reviewed in accordance with the change control program described in TS 6.5.2. The deletion and relocation of Table 3.6.4-1 complies with the guidance contained in GL 91-08. The staff finds the proposed changes acceptable.

(27) Delete Footnote (#) associated with SR 4.6.4.2.

This footnote provides an extension of the test interval until the first refueling outage, which occurred in 1987, and is no longer necessary. The proposed change is editorial in nature and does not result in a technical change to the current requirements. The staff finds the proposed change acceptable.

(28) The proposed change revises SR 4.6.4.1 and SR 4.6.4.2 to clarify that the various requirements relate to "primary containment or drywell" isolation valves. Inserting the word "primary containment or drywell" in the SRs clarifies the SRs and makes the wording consistent with that of the LCO and ACTION Statements.

These changes are provided for clarification only and do not result in any technical change as these requirements have always been understood to relate to "primary containment or drywell" isolation valves. The staff finds the proposed change acceptable.

(29) LCO 3.6.5.3, ACTION Statement 3.6.5.3, and SR 4.6.5.3 are proposed to be revised to delete the references to Table 3.6.5.3-1 (Secondary Containment Ventilation System Automatic Isolation Dampers) and to delete the reference to isolation times. Table 3.6.5.3-1 denotes secondary containment ventilation system automatic isolation damper functions, maximum isolation times, damper isolation groups and applicable operational conditions. The table is being deleted from the TSs and relocated to the TRM (as discussed below). The licensee had originally proposed to include the word "required" in front of the phrase "secondary containment ventilation system isolation dampers." The word "required" was also proposed in other technical specifications as a means to preserve the specified modes in the TSs rather than in the table to be relocated. Since this language was not specific in several proposed TS changes, the licensee agreed to remove all the phrases "required." However, in the November 10, 1994, letter, the license overlooked this one TS. In the February 8, 1995, letter, the use of the word "required" is proposed to be removed from this TS. This applies to the LCO, Action Statement, and SR where the word "required" was initially proposed. See item (30) and (31) below.

The details in this table are for information only except for the listing of the applicable operational conditions for the associated dampers (as discussed below). Since dampers may not be required to be OPERABLE in each of the operational conditions listed in the proposed change to the APPLICABILITY statement, the words in the LCO, ACTION Statement, and Surveillance Requirement have been revised. With this proposed change, the list of secondary containment ventilation system automatic isolation dampers is not necessary to support the actions or surveillances required by the TSs. The proposed changes are consistent with the guidance in GL 91-08. The staff finds the proposed changes acceptable.

(30) The APPLICABILITY statement for LCO 3.6.5.3 is proposed to be revised to replace the reference to Table 3.6.5.3-1 (Secondary Containment Ventilation System Automatic Isolation Dampers) with OPERATIONAL CONDITIONS 1, 2, 3, and ##. Table 3.6.5.3-1 denotes secondary containment ventilation system automatic isolation damper functions, maximum isolation times, damper isolation groups and applicable operational conditions. The table is being deleted from the TSs and relocated to the TRM (as discussed below). The APPLICABILITY statement was proposed to be changed concurrent with the change adding the word "required", as noted above. However, this use of the word "required" has been withdrawn by the licensee; see item (29) above. In reviewing the table to be removed, it was determined that applicability statements or conditions were included for previous convenience. In keeping all applicability statements in the TS with relocation of table information to the TRM, certain fuel building damper statements needed to be

addressed. The staff has corrected the proposed statements consistent with the guidance of GL 91-08 and changed the statement to read:

- OPERATING CONDITIONS; 1, 2, and 3
- FOR FUEL BUILDING DAMPERS; OPERATING CONDITIONS 1, 2, 3 AND ##.

This assures the dampers needed for fuel movement will be Operable whenever fuel is moved and that the requirements for operability have been adequately moved from the table to the LCO. The licensee's language was inadequate and would have required the dampers operable in all modes which is beyond the requirements of the original TS wording and the purpose of GL 91-08. The licensee agrees to this clarification of their proposal.

With the proposed incorporation of the operational conditions into the APPLICABILITY statement, the details in Table 3.6.5.3-1 are not necessary to support the actions or the surveillances required by the TSs. The proposed change is consistent with the guidance provided in GL 91-08 and does not result in a technical change to the TSs. The staff finds the proposed changes acceptable.

(31) SR 4.6.5.3.a is proposed to be revised to delete the word "specified" since, with the deletion of Table 3.6.5.3-1 "Secondary Containment Ventilation System Automatic Isolation Dampers", the isolation times for automatic dampers are no longer specified. Table 3.6.5.3-1 is being deleted from the Tss and relocated to the TRM (as discussed below). The use of the word "required" was removed by the licensee; see item (29) above.

This is an editorial change associated with the relocation of Table 3.6.5.3-1 to the TRM. The staff finds the proposed change acceptable.

(32) Table 3.6.5.3-1 and associated footnotes are proposed to be deleted from the Tss and relocated to the TRM. Table 3.6.5.3-1 currently denotes secondary containment ventilation system automatic isolation dampers. The table also denotes a list of damper functions, maximum isolation times, damper group numbers (associated with Specification 3/4.3.2, "Isolation Actuation Instrumentation") and applicable operational conditions for the associated dampers.

With the proposed changes to TS 3/4.6.5.3, the requirements apply in general to each secondary containment ventilation system automatic isolation damper. The information being relocated to the TRM is controlled and subsequent changes will be reviewed in accordance with the change control program described in Specification 6.5.2. As a result, the deletion and subsequent relocation of this information from the TSs complies with the guidance contained in GL 91-08. The proposed changes are acceptable to the staff.

(33) Delete Footnote (###) associated with SR 4.6.5.3.b.

This footnote provides an extension of the test interval until the first refueling outage, which occurred in 1987. The footnote action has been completed and the footnote is no longer necessary. The proposed change is editorial in nature and does not result in a technical change to the current requirements. The staff finds the proposed change acceptable.

(34) SR 4.6.5.3.b is being revised to clarify that the containment isolation test signal identified in this surveillance is a secondary containment isolation signal (versus primary containment isolation signal, for example) and that the isolation dampers to be verified are secondary containment automatic isolation dampers.

This proposed change provides clarification only, does not alter the intent of the surveillance, and does not result in any technical change to the operability or testing requirements. The staff finds the proposed change acceptable.

(35) LCO 3.8.4.1, ACTION Statement 3.8.4.1, and SR 4.8.4.1 are proposed to be revised to delete the references to Table 3.8.4.1-1 (Primary Containment Penetration Conductor Overcurrent Protection Devices). The referenced Table 3.8.4.1-1 is being deleted from the Tss and relocated to the TRM (as discussed below).

The LCO is also being revised to state that the scope of this TS includes each of the primary and backup overcurrent protective devices associated with each primary containment electrical penetration circuit, but excludes those circuits for which credible fault currents would not exceed the electrical penetrations' design ratings. This statement reflects the basis for which the contents of the table were originally developed and does not constitute a technical change. This revision to the LCO is also as recommended in GL 91-08. The staff finds the proposed changes acceptable.

(36) Table 3.8.4.1-1 is proposed to be deleted from the TSs and relocated to the TRM. Table 3.8.4.1-1 currently denotes primary containment penetration conductor overcurrent protection devices for 4.16 kV circuit breakers, 120/140 VAC molded case circuit breakers, 480 VAC molded case circuit breakers, and air circuit breakers. For each type of circuit breaker, the table denotes the equipment or device number and location of each circuit breaker.

With the proposed revisions to TS 3/4.8.1, the requirements apply in general to the primary and backup overcurrent protection device associated with each primary containment electrical penetration circuit, excluding those circuits for which credible fault currents would not exceed the electrical penetration's design rating. The information in Table 3.8.4.1-1 is being relocated to the TRM. This information is controlled and subsequent changes will be reviewed in accordance with the change control program described in Specification 6.5.2. In addition, USAR Section 8.3.1.4.2 addresses the requirements for overcurrent protection of the containment electrical penetrations. As a result, the deletion and subsequent relocation of this information from the TSs complies with the guidance contained in GL 91-08. The staff finds the proposed changes acceptable.

(37) LCO 3.8.4.2 and ACTION Statement 3.8.4.2 are proposed to be revised to delete the references to Table 3.8.4.2-1 (Other Overcurrent Protective Devices). The referenced Table 3.8.4.2-1 is being deleted from the TSs and relocated to the TRM (as discussed below).

This LCO is also being revised to state that the scope of the TS includes each primary overcurrent protection device for the Main Control Room safety-related lighting and the primary and secondary RPS Alternate Source of Power. This statement reflects the basis for which the contents of the table were originally developed and does not constitute a technical change. The staff finds the proposed changes acceptable.

(38) Table 3.8.4.2-1 is proposed to be deleted from the TSs and relocated to the TRM. Table 3.8.4.2-1 currently denotes other overcurrent protective devices for Main Control Room lighting and Reactor Protection System alternate source of power. For each type of overcurrent protective device, the table denotes the equipment or device number.

With the proposed changes to TS 3/4.8.4.2, the information in Table 3.8.4.2-1 is being relocated to the TRM. This information is controlled and subsequent changes will be reviewed in accordance with the change control program described in Specification 6.5.2. As a result, the deletion and subsequent relocation of this information from the TSs complies with the guidance contained in GL 91-08. With the proposed change, the requirements apply to each required overcurrent protection device and the information in the table is not necessary to support the actions or surveillances required by the TSs. The staff finds the proposed changes acceptable.

(39) LCO 3.8.4.4, ACTION Statement 3.8.4.4, and SR 4.8.4.4 are proposed to be revised to delete the references to the component list currently included in the LCO. The component list denotes the equipment identification number, device number, and location of A.C. circuits inside containment. The component list is being relocated to the TRM and is controlled and subsequent changes will be reviewed in accordance with the change control program described in Specification 6.5.2.

The LCO is also being revised to state that each 480V and 240/120V A.C. circuit inside containment for the containment Building HVAC, Drywell Cooling HVAC, RWCU, Inclined Fuel Transfer Tube, and Reactor Building Main Hoist Systems without redundant penetration protection shall be de-energized. This statement reflects the basis for which the contents of the table were originally developed and does not constitute a technical change. The deletion and subsequent relocation of this information from the TS complies with the guidance contained in GL 91-08. The staff finds the proposed changes acceptable.

(40) Revisions to TS 6.5.2, "Technical Review and Control," are proposed to add a reference to the Technical Requirements Manual and to require that implementing procedures and changes to the manual be approved in accordance with TS 6.5.2.1 and that records of these activities be kept.

The staff finds the proposed changes acceptable.

(41) A revision to TS 6.8.1, "Procedures and Programs," is proposed to add the Technical Requirements Manual to the list of documents subject to the controls of TS 6.8.1.

The staff finds the proposed changes acceptable.

The staff's review of the proposed changes determined that the relocation of Tables 3.4.3.2-1, 3.4.3.2-2, 3.6.1.3-1, 3.6.4-1, 3.6.5.3-1, 3.8.4.1-1, and 3.8.4.1-2 does not eliminate the requirements for the licensee to ensure that associated components are capable of performing their safety function. Although the tables listed above are relocated from the technical specifications to the Technical Requirements Manual, the licensee must evaluate any changes to the components listed in the tables in accordance with 10 CFR 50.59. Should the licensee's evaluation conclude that an unresolved safety question is involved, due to either (1) an increase in the probability or consequences of accidents or malfunctions of equipment important to safety, (2) the creation of a possibility for an accident or malfunction of a different type than any evaluated previously, or (3) a reduction in the margin of safety, NRC approval and a license amendment would be required prior to implementation of the change. NRC inspection and enforcement programs also enable the staff to monitor facility changes and licensee adherence to commitments and to take remedial action that may be appropriate.

The staff's review concluded that 10 CFR 50.36 does not require Tables 3.4.3.2-1, 3.4.3.2-2, 3.6.1.3-1, 3.6.4-1, 3.6.5.3-1, 3.8.4.1-1, or 3.8.4.1-2 to be retained in the TSs. Requirements related to the operability, applicability, and surveillance requirements, including performance of testing to ensure operability of the listed components is retained due to the components' importance in mitigating the consequences of an accident. However, the staff determined that the inclusion of these tables are an operational detail related to the licensee's safety analyses which are adequately controlled by the requirements of 10 CFR 50.59. Therefore, the continued processing of license amendments related to revisions to these tables, where the revisions to those requirements do not involve an unreviewed safety question under 10 CFR 50.59, would afford no significant benefit with regard to protecting the public health and safety.

The staff has concluded, therefore, that relocation of Tables 3.4.3.2-1, 3.4.3.2-2, 3.6.1.3-1, 3.6.4-1, 3.6.5.3-1, 3.8.4.1-1, and 3.8.4.1-2 is acceptable because (1) their inclusion in the TSs is not specifically required by 10 CFR 50.36 or other regulations, (2) the tables have been relocated to the Technical Review Manual, are adequately controlled by TS 6.5.2 and 10 CFR 50.59, (3) their inclusion in the TSs is not required to avert an immediate threat to the public health and safety, and (4) changes that are deemed to involve an unreviewed safety question, will require prior NRC approval in accordance with 10 CFR 50.59(c).

4.0 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 comment.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant changes 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 65815). 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.

6.0 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.

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