Docket No. 50-416 December 3, 1992

Mr. William T. Cottle
Vice President, Operations GGNS
Entergy Operations, Inc.
Post Office Box 756
Port Gibson, Mississippi 39150

Dear Mr. Cottle:

SUBJECT: ISSUANCE OF AMENDMENT NO. 102 TO FACILITY OPERATING LICENSE NO. NPF-29 - GRAND GULF NUCLEAR STATION, UNIT 1 (TAC NO. M84500)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 102 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. This amendment revises the Technical Specifications (TS) in response to your application dated September 14, 1992.

The amendment changes the TS by removing Table 3.6.4-1, "Containment and Drywell Isolation Valves"; Table 3.6.6.2-1, "Secondary Containment Ventilation System Automatic Isolation Dampers/Valves"; Table 3.8.4.1-1, "Primary Containment Penetration Conductor Overcurrent Protective Devices"; and Table 3.8.4.2-1, "Motor Operated Valve Thermal Overload Protection." Table 3.3.2-1, "Isolation Actuation Instrumentation," is modified as a result of the table deletions. The lists removed from the TS will be incorporated into plant procedures subject to the administrative controls of TS 6.8 and 6.5.3. Related Definitions and Bases are modified to be consistent with the table deletions. Guidance on the proposed TS changes was provided by Generic Letter 91-08, dated May 6, 1991.

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

Sincerely,
ORIGINAL SIGNED BY:
Paul W. O'Connor, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 102 to NPF-29
2. Safety Evaluation

cc w/enclosures:
See next page

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

December 3, 1992

Docket No. 50-416

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Sincerely,

Paul W. O'Connor

Paul W. O'Connor, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.102 to NPF-29
2. Safety Evaluation

cc w/enclosures:
See next page

cc:

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

ENTERGY OPERATIONS, INC.

SYSTEM ENERGY RESOURCES, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

MISSISSIPPI POWER AND LIGHT COMPANY

DOCKET NO. 50-416

GRAND GULF NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 102
License No. NPF-29

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated September 14, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

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

2. Accordingly, 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-29 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 102, are hereby incorporated into this license. Entergy Operations, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John T. Larkins

John T. Larkins, Director
Project Directorate IV-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 3, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 102

FACILITY OPERATING LICENSE NO. NPF-29

DOCKET NO. 50-416

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

REMOVE PAGES	INSERT PAGES
1-2	1-2
1-6	1-6
1-7	1-7
3/4 3-10 thru 3/4 3-15	3/4 3-10 thru 3/4 3-15
3/4 3-21	3/4 3-21
3/4 6-1 thru 3/4 6-4	3/4 6-1 thru 3/4 6-4
3/4 6-13	3/4 6-13
3/4 6-28 thru 3/4 6-45	3/4 6-28 thru 3/4 6-30
3/4 6-49 thru 3/4 6-54	3/4 6-49 thru 3/4 6-50
3/4 8-19 thru 3/4 8-53	3/4 8-19 thru 3/4 8-21
–	3/4 8-46
–	3/4 8-47
B 3/4 6-7	B 3/4 6-7
B 3/4 6-8a	B 3/4 6-8a
B 3/4 8-3	B 3/4 8-3

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AVERAGE PLANAR EXPOSURE

1.2 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is tested.

CORE ALTERATION

1.7 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Normal movement of the SRMs, IRMs, LPRMs, TIPs, or special movable detectors is not considered to be CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CRITICAL POWER RATIO

1.8 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the ANFB correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.9 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

DRYWELL INTEGRITY

1.10 DRYWELL INTEGRITY shall exist when:

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

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

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are opened under administrative control as permitted by Specification 3.6.4.

b. The drywell equipment hatch is closed and sealed.

c. The drywell airlock is in compliance with the requirements of Specification 3.6.2.3.

d. The drywell leakage rates are within the limits of Specification 3.6.2.2.

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

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

MINIMUM CRITICAL POWER RATIO

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

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.26 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.3 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semiannual Radioactive Effluent Release Reports required by Specifications 6.9.1.6, 6.9.1.7, 6.9.1.8 and 6.9.1.9.

OPERABLE - OPERABILITY

1.27 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

1.28 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

1.29 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

1.30 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

1.31 PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All 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 for valves that are opened under administrative control as permitted by Specification 3.6.4.

b. The containment equipment hatch is closed and sealed.

c. Each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The containment leakage rates are within the limits of Specification 3.6.1.2.

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)

1.32 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that 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 Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.33 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.34 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3833 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.35 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE EVENT

1.36 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.37 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

1.38 SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All Auxiliary Building and Enclosure Building penetrations required to be closed during accident conditions are either:

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

    2. Closed by at least one manual valve, blind flange, rupture disc or deactivated automatic valve or damper, as applicable, secured in its closed position.

b. All Auxiliary Building and Enclosure Building equipment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.6.3.

d. The door in each access to the Auxiliary Building and Enclosure Building is closed, except for normal entry and exit.

e. The sealing mechanism associated with each Auxiliary Building and Enclosure Building penetration, e.g., welds, bellows or O-rings, is OPERABLE.

SHUTDOWN MARGIN

1.39 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

SITE BOUNDARY

1.40 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

1.41 DELETED

1.42 DELETED

STAGGERED TEST BASIS

1.43 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

1.44 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNIDENTIFIED LEAKAGE

1.45 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

UNRESTRICTED AREA

1.46 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of MEMBERS OF THE PUBLIC from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

1.47 DELETED

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	2	1, 2, 3 and #	20
b. Reactor Vessel Water Level- Low Low Level 2 (ECCS - Division 3)	4	1, 2, 3 and #	29
c. Reactor Vessel Water Level- Low Low Low, Level 1 (ECCS - Division 1 and Division 2)	2	1, 2, 3 and #	29
d. Drywell Pressure - High***	2	1, 2, 3	20
e. Drywell Pressure-High (ECCS - Division 1 and Division 2)	2	1, 2, 3	29
f. Drywell Pressure-High (ECCS - Division 3)	4	1, 2, 3	29
g. Containment and Drywell Ventilation Exhaust Radiation - High High	2(b)	1, 2, 3 and *	21
h. Manual Initiation	2	1, 2, 3 and *#	22
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***	2	1, 2, 3	23
c. Main Steam Line Pressure - Low	2	1	24
d. Main Steam Line Flow - High	8	1, 2, 3	23
e. Condenser Vacuum - Low	2	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
2. MAIN STEAM LINE ISOLATION (Continued)			
f. Main Steam Line Tunnel Temperature - High	2	1, 2, 3	23
g. Main Steam Line Tunnel Δ Temp.- High	2	1, 2, 3	23
h. Manual Initiation	2	1, 2, 3	22
3. SECONDARY CONTAINMENT ISOLATION			
a. Reactor Vessel Water Level-Low Low, Level 2	2	1, 2, 3, and #	25
b. Drywell Pressure - High***	2	1, 2, 3	25
c. Fuel Handling Area Ventilation Exhaust Radiation - High High	2	1, 2, 3, and *	25
d. Fuel Handling Area Pool Sweep Exhaust Radiation - High High	2	1, 2, 3, and *	25
e. Manual Initiation	2 2	1, 2, 3 *	26 25
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/room	1, 2, 3	27
d. Equipment Area Δ Temp. - High	1/room	1, 2, 3	27
e. Reactor Vessel Water Level - Low Low, Level 2	2	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)			
f. Main Steam Line Tunnel Ambient Temperature - High	1	1, 2, 3	27
g. Main Steam Line Tunnel Δ Temp. - High	1	1, 2, 3	27
h. SLCS Initiation	1	1, 2, 5##	30
i. Manual Initiation	2	1, 2, 3	26
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION			
a. RCIC Steam Line Flow - High			
1. Pressure	1	1, 2, 3	27
2. Time Delay	1	1, 2, 3	27
b. RCIC Steam Supply Pressure - Low	1	1, 2, 3	27
c. RCIC Turbine Exhaust Diaphragm Pressure - High	2	1, 2, 3	27
d. RCIC Equipment Room Ambient Temperature - High	1	1, 2, 3	27
e. RCIC Equipment Room Δ Temp. - High	1	1, 2, 3	27
f. Main Steam Line Tunnel Ambient Temperature - High	1	1, 2, 3	27
g. Main Steam Line Tunnel Δ Temp. - High	1	1, 2, 3	27
h. Main Steam Line Tunnel Temperature Timer	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			
i. RHR Equipment Room Ambient Temperature - High	1/room	1, 2, 3	27
j. RHR Equipment Room Δ Temp. - High	1/room	1, 2, 3	27
k. RHR/RCIC Steam Line Flow - High	1	1, 2, 3	27
l. Manual Initiation	1	1, 2, 3	26
m. Drywell Pressure-High (ECCS-Division 1 and Division 2)	1	1, 2, 3	27
6. RHR SYSTEM ISOLATION			
a. RHR Equipment Room Ambient Temperature - High	1/room	1, 2, 3	28
b. RHR Equipment Room Δ Temp. - High	1/room	1, 2, 3	28
c. Reactor Vessel Water Level - Low, Level 3***	2	1, 2, 3	28
	2(c)	4, 5	31
d. Reactor Vessel (RHR Cut-in Permissive) Pressure - High***	2	1, 2, 3	28
e. Drywell Pressure - High***	2	1, 2, 3	28
f. Manual Initiation	2	1, 2, 3	26

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION
ACTION

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

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

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

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

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

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

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

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

ACTION 28 - Within one hour lock the affected system isolation valves closed, or verify, by remote indication, that the valve is closed and electrically disarmed, or isolate the penetration(s) and declare the affected system inoperable.

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

ACTION 30 - Declare the affected SLCS pump inoperable.

ACTION 31 - Isolate the shutdown cooling common suction line within one hour if it is not needed for shutdown cooling or initiate action within one hour to establish SECONDARY CONTAINMENT INTEGRITY.

NOTES

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

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

*** Trip function commom to RPS Instrumentation.

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

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

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

NOTES (Continued)

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

(c) Only required to isolate RHR system isolation valves E12-F008 and E12-F009. One trip system and/or isolation valve may be inoperable for up to 14 days without placing the trip system in the tripped condition provided the diesel generator associated with the OPERABLE isolation valve is OPERABLE.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -41.6 inches *	≥ -43.8 inches
b. Reactor Vessel Water Level- Low Low, Level 2 (ECCS - Division 3)	≥ -41.6 inches*	≥ -43.8 inches
c. Reactor Vessel Water Level- Low Low Low, Level 1 (ECCS Division 1 and Division 2)	≥ -150.3 inches*	≥ -152.5 inches
d. Drywell Pressure - High	≤ 1.23 psig	≤ 1.43 psig
e. Drywell Pressure-High (ECCS - Division 1 and Division 2)	≤ 1.39 psig	≤ 1.44 psig
f. Drywell Pressure-High (ECCS - Division 3)	≤ 1.39 psig	≤ 1.44 psig
g. Containment and Drywell Ventilation Exhaust Radiation - High High	≤ 3.6 mR/hr**	≤ 4.0 mR/hr**
h. Manual Initiation	NA	NA
2. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -150.3 inches*	≥ -152.5 inches
b. Main Steam Line Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full power background
c. Main Steam Line Pressure - Low	≥ 849 psig	≥ 837 psig
d. Main Steam Line Flow - High	≤ 169 psid	≤ 176.5 psid
e. Condenser Vacuum - Low	≥ 9 inches Hg. Vacuum	≥ 8.7 inches Hg. Vacuum
f. Main Steam Line Tunnel Temperature - High	≤ 185°F**	≤ 191°F**

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Flow - High	$\leq 10^{(a)\#\#\#}$
b. RCIC Steam Supply Pressure - Low	$\leq 10^{(a)}$
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
d. RCIC Equipment Room Ambient Temperature - High	NA
e. RCIC Equipment Room Δ Temp. - High	NA
f. Main Steam Line Tunnel Ambient Temp. - High	NA
g. Main Steam Line Tunnel Δ Temp. - High	NA
h. Main Steam Line Tunnel Temperature Timer	NA
i. RHR Equipment Room Ambient Temperature - High	NA
j. RHR Equipment Room Δ Temp. - High	NA
k. RHR/RCIC Steam Line Flow - High	NA
l. Manual Initiation	NA
m. Drywell Pressure - High (ECCS Division 1 and Division 2)	$\leq 10^{(a)}$
6. RHR SYSTEM ISOLATION	
a. RHR Equipment Room Ambient Temperature - High	NA
b. RHR Equipment Room Δ Temp. - High	NA
c. Reactor Vessel Water Level - Low, Level 3	$\leq 10^{(a)}$
d. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA
e. Drywell Pressure - High	NA
f. Manual Initiation	NA

(a) The isolation system instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Isolation system instrumentation response time specified includes the delay for diesel generator starting assumed in the accident analysis.

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

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

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

***Isolation system instrumentation response time for air operated dampers. No diesel generator delays assumed.

#Isolation system instrumentation response time specified for the Trip Function actuating each valve shall be added to the valve's required isolation time to obtain the valve's ISOLATION SYSTEM RESPONSE TIME.

###Includes time delay of 3 to 7 seconds.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2	S	Q	R(c)	1, 2, 3 and #
b. Reactor Vessel Water Level-Low Low, Level 2 (ECCS - Division 3)	S	Q	R(c)	1, 2, 3 and #
c. Reactor Vessel Water Level-Low Low Low, Level 1 (ECCS - Division 1 and Division 2)	S	Q	R(c)	1, 2, 3 and #
d. Drywell Pressure - High	S	Q	R(c)	1, 2, 3
e. Drywell Pressure-High (ECCS - Division 1 and Division 2)	S	Q	R(c)	1, 2, 3
f. Drywell Pressure-High (ECCS - Division 3)	S	Q	R(c)	1, 2, 3
g. Containment and Drywell Ventilation Exhaust Radiation - High High	S	Q(a)	A	1, 2, 3 and *
h. Manual Initiation	NA	Q(a)	NA	1, 2, 3 and *#
2. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	R(c)	1, 2, 3
b. Main Steam Line Radiation - High	S	Q	R	1, 2, 3
c. Main Steam Line Pressure - Low	S	Q	R(c)	1
d. Main Steam Line Flow - High	S	Q	R(c)	1, 2, 3
e. Condenser Vacuum - Low	S	Q	R(c)	1, 2**, 3**

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at P_a, 11.5 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 La.

b. At least once per 31 days by verifying that all containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that are opened under administrative control as permitted by Specification 3.6.4.

c. By verifying each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. By verifying the suppression pool is in compliance with the requirements of Specification 3.6.3.1.

*See Special Test Exception 3.10.1

**Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, steam tunnel or drywell and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed more often than once per 92 days.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, 0.437 percent by weight of the containment air per 24 hours at P_a, 11.5 psig.

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and all valves# subject to Type B and C tests when pressurized to P_a, 11.5 psig.

c. Less than or equal to 100 scf per hour for all four main steam lines through the isolation valves when tested at P_a, 11.5 psig.

d. 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 1.10 P_a, 12.65 psig.

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

ACTION:

With:

a. The measured overall integrated containment leakage rate exceeding 0.75 L_a, or

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

c. The measured leakage rate exceeding 100 scf per hour for all four main steam lines through the isolation valves, or

d. 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 or equal to 0.75 L_a, and

#Except for those that are hydrostatically leak tested.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

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

c. The leakage rate to less than 100 scf per hour for all four main steam lines through the isolation valves, and

d. 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

4.6.1.2 The 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 P_a, 11.5 psig, during each 10-year service period.

b. If any periodic Type A test fails to meet 0.75 L_a, 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 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a, 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 containment leakage rate, L'_v, calculated in accordance with ANSI N-45.4-1972, Appendix C, is within 25 percent of the containment leakage rate, L_v, measured prior to the introduction of the superimposed leak.

   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 containment or bled from the containment during the supplemental test to be between 0.75 L_a and 1.25 L_a.

#Except for those that are hydrostatically leak tested.

*The third Type A test within the first 10-year service period shall be conducted prior to startup following the sixth refueling outage. This is an exemption from 10 CFR Part 50, Appendix J Requirements.

SURVEILLANCE REQUIREMENTS (Continued)

d. Type B and C tests shall be conducted with gas at P_a, 11.5 psig,* at intervals no greater than 24 months except for tests involving:

   1. Air locks,

   2. Main steam line isolation valves,

   3. Penetrations using continuous leakage monitoring systems,

   4. Valves pressurized with fluid from a seal system,

   5. Containment isolation valves in hydrostatically tested lines which penetrate the primary containment, and

   6. Purge supply and exhaust isolation valves with resilient material seals.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. Main steam line isolation valves shall be leak tested at least once per 18 months.

g. Type B tests for penetrations employing a continuous leakage monitoring system shall be conducted at P_a, 11.5 psig, at intervals no greater than once per 3 years.

h. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a, 12.65 psig, and the seal system capacity is adequate to maintain system pressure for at least 30 days.

i. Containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at least once per 18 months.

j. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.9.2.

k. The provisions of Specification 4.0.2 are not applicable to Specifications 4.6.1.2.a, 4.6.1.2.b, 4.6.1.2.c, 4.6.1.2.d, 4.6.1.2.e, and 4.6.1.2.g.

*Unless a hydrostatic test is required.

CONTAINMENT SYSTEMS

3/4.6.2 DRYWELL

DRYWELL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.2.1 DRYWELL INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

Without DRYWELL INTEGRITY, restore DRYWELL INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 DRYWELL INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all drywell penetrations** not capable of being closed by OPERABLE drywell automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that are opened under administrative control as permitted by Specification 3.6.4.

b. By verifying the drywell air lock is in compliance with the requirements of Specification 3.6.2.3.

c. By verifying the suppression pool is in compliance with the requirements of Specification 3.6.3.1.

*See Special Test Exception 3.10.1.

**Except valves, blind flanges, and deactivated automatic valves which are located inside the drywell or containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed more often than once per 92 days.

CONTAINMENT SYSTEMS

DRYWELL BYPASS LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.2.2 Drywell bypass leakage shall be less than or equal to 10% of the acceptable $A/\sqrt{k}$ design value of 0.90 ft^2.

APPLICABILITY: When DRYWELL INTEGRITY is required per Specification 3.6.2.1.

ACTION:

With the drywell bypass leakage greater than 10% of the acceptable $A/\sqrt{k}$ design value of 0.90 ft^2, restore the drywell bypass leakage to within the limit prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The drywell bypass leakage rate test shall be conducted at least once per 18 months at an initial differential pressure of 3.0 psid and the $A/\sqrt{k}$ shall be calculated from the measured leakage. One drywell airlock door shall remain open during the drywell leakage test such that each drywell door is leak tested during at least every other drywell leakage rate test.

a. If any drywell bypass leakage rate test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the limit, a test shall be performed at least once every 9 months until two consecutive tests meet the limit, at which time the above test schedule may be resumed.

b. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.2.3.

c. The provisions of Specification 4.0.2 are not applicable.

3/4.6.4 CONTAINMENT AND DRYWELL ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With one or more of the 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,* or

c. Isolate each affected penetration by use of at least one closed manual valve or blind flange*.

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

*Isolation valves, except MSIVs, closed to satisfy these requirements may be reopened on an intermittent basis under administrative controls. OPERATIONAL CONDITION changes, as provided by Specification 3.0.4, are not allowed while isolation valves are open under these administrative controls.

#Isolation valves are also required to be OPERABLE when their associated actuation instrumentation is required to be OPERABLE per Table 3.3.2-1.

**Except for E12-F008 and E12-F009 in OPERATIONAL CONDITIONS 4 and 5 take action per Specification 3.3.2, Table 3.3.2-1, Trip Function 6.c.

***Normally closed or locked closed manual valves may be opened on an intermittent basis under administrative control.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each 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 applicable isolation time.

4.6.4.2 Each automatic 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.

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

4.6.4.4 [DELETED]

##The provisions of Specification 4.0.4 are not applicable to automatic main steam line valves for entry into OPERATIONAL CONDITIONS 2 or 3, provided the surveillance is performed within 12 hours after reaching a reactor steam pressure of 600 psig and prior to entry into OPERATIONAL CONDITION 1.

Pages 3/4 6-30 through 3/4 6-45 have been Intentionally Deleted

3/4.6.5 DRYWELL VACUUM RELIEF

LIMITING CONDITION FOR OPERATION

3.6.5 Both drywell post-LOCA vacuum relief subsystems and both drywell purge vacuum relief subsystems shall be OPERABLE with associated vacuum breakers and isolation valves closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one of the drywell post-LOCA vacuum relief subsystems and/or one of the drywell purge vacuum relief subsystems inoperable for opening but known to be closed, restore the inoperable subsystem(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With two of the post-LOCA vacuum relief subsystems inoperable for opening but known to be closed, provided that both of the drywell purge vacuum relief subsystems are OPERABLE, restore the inoperable subsystems to OPERABLE status within 30 days or be in a least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With two of the post-LOCA vacuum relief subsystems and one of the drywell purge vacuum relief subsystems inoperable for opening but known to be closed, restore one inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one of the drywell isolation vacuum breakers open, restore the open vacuum breaker to the closed position within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With the position indicator of an OPERABLE drywell vacuum breaker or associated isolation valve of the drywell vacuum relief subsystems inoperable, verify the vacuum breaker or isolation valve to be closed at least once per 24 hours by local indication. Otherwise be in a least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.5 Each post-LOCA and purge system vacuum breaker and associated isolation valve shall be:

a. Verified closed at least once per 7 days.

SECONDARY CONTAINMENT AUTOMATIC ISOLATION DAMPERS/VALVES

LIMITING CONDITION FOR OPERATION

3.6.6.2 Each secondary containment ventilation system automatic isolation damper/valve shall be OPERABLE.

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

ACTION:

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

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

b. Isolate each affected penetration by use of at least one deactivated automatic damper/valve secured in the isolation position, or

c. Isolate each affected penetration by use of at least one closed manual valve or blind flange.

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 primary or secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.6.2 Each secondary containment ventilation system automatic isolation damper/valve shall be demonstrated OPERABLE:

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

b. During COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each isolation damper/valve actuates to its isolation position.

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

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

Pages 3/4 6-50 through 3/4 6-54 have been Intentionally 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 Primary and backup containment penetration conductor overcurrent protective devices (i.e., circuit breakers) associated with each primary containment electrical penetration circuit shall be OPERABLE. The scope of these protective circuit breakers excludes those circuits for which credible fault currents would not exceed the electrical penetration design rating.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more of the primary or backup containment penetration conductor overcurrent protective devices inoperable, declare the affected system or component inoperable and apply the appropriate ACTION statement for the affected system, and:

   1. For 6.9 kV circuit breakers, de-energize the 6.9 kV circuit(s) by tripping the associated redundant circuit breaker(s) within 72 hours and verify the redundant circuit breaker to be tripped at least once per 7 days thereafter.
   2. For 480 volt circuit breakers, remove the inoperable circuit breaker(s) from service by racking out the breaker within 72 hours and verify the inoperable breaker(s) to be racked out at least once per 7 days thereafter.

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

SURVEILLANCE REQUIREMENTS

4.8.4.1 Each of the primary and backup containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE:

a. At least once per 18 months:

   1. By verifying that the medium voltage 6.9 kV circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the 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 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.

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 a current in excess of 120% of the breakers nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to insure 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 of the affected equipment. 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.

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

Pages 3/4 8-21 through 3/4 8-45 have been Intentionally Deleted.

(Next page is 3/4 8-46)

TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

(e) 208/120 VAC Circuit Breakers (Continued)
Westinghouse BA

BREAKER NUMBER	TIME O.C. PICKUP (Amperes)	RESPONSE TIME (Seconds)	SYSTEM/COMPONENT AFFECTED
1H22-P110A-1	15	4.0	HEAT TRACE FOR SLCS N1R63T321A
1H22-P110A-4	15	4.0	HEAT TRACE FOR SLCS N1R63T322A
1H22-P110A-5	15	4.0	HEAT TRACE FOR SLCS N1R63T323A
1H22-P110A-8	15	4.0	HEAT TRACE FOR SLCS N1R63T324A
1H22-P110B-1	15	4.0	HEAT TRACE FOR SLCS N1R63T321B
1H22-P110B-4	15	4.0	HEAT TRACE FOR SLCS N1R63T322B
1H22-P110B-5	15	4.0	HEAT TRACE FOR SLCS N1R63T323B
1H22-P110B-8	15	4.0	HEAT TRACE FOR SLCS N1R63T324B

*3 Pole Breaker

ELECTRICAL POWER SYSTEMS

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection of each valve whose motor operator performs a safety function shall be OPERABLE or shall be bypassed either continuously or only under accident conditions, as applicable, by an OPERABLE bypass device.

APPLICABILITY: Whenever the motor operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves not OPERABLE or not bypassed either continuously or only under accident conditions, as applicable, bypass the thermal overload within 8 hours or declare the affected valve(s) inoperable and apply the appropriate ACTION statement(s) for the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2.1 The thermal overload protection which is bypassed either continuously or only under accident conditions for the above required valves shall be verified to be bypassed continuously or only under accident conditions, as applicable, by an OPERABLE bypass device (1) by the performance of a CHANNEL FUNCTIONAL TEST of the bypass circuitry for those thermal overloads which are normally in force during plant operation and bypassed under accident conditions and (2) by verifying that the thermal overload protection is bypassed for those thermal overloads which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing:

a. For those thermal overloads which are normally in force during plant operation and bypassed under accident conditions:

   1. At least once per 92 days for the individual valve bypass circuitry.

   2. At least once per 18 months for the ECCS portion of the channel.

b. At least once per 18 months for those thermal overloads which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing.

c. Following maintenance on the motor starter.

4.8.4.2.2 The thermal overload protection which is not bypassed for the above required valves shall be demonstrated OPERABLE at least once per 18 months by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of all thermal overloads for the above required valves.

4.8.4.2.3 The thermal overload protection for the above required valves which is continuously bypassed and temporarily placed in force only when the valve motor is undergoing periodic or maintenance testing shall be verified to be bypassed following periodic or maintenance testing during which the thermal overload protection was temporarily placed in force.

Pages 3/4 8-47 through 3/4 8-53 have been Intentionally Deleted.

REACTOR PROTECTION SYSTEM ELECTRIC POWER MONITORING

LIMITING CONDITION FOR OPERATION

3.8.4.3 Two RPS electric power monitoring assemblies for each inservice RPS MG set or alternate power supply shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one RPS electric power monitoring assembly for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable power monitoring system 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 assemblies for an inservice RPS MG set or alternate power supply inoperable, restore at least one electric power monitoring assembly 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 assemblies shall be determined OPERABLE:

a. By performance of a CHANNEL FUNCTIONAL TEST each time the plant is in COLD SHUTDOWN for a period of more than 24 hours, unless performed in the previous 6 months.

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	Bus A	$\leq$ 132.9 VAC
		Bus B	$\leq$ 133.0 VAC
2.	Under-voltage	Bus A	$\geq$ 115.0 VAC
		Bus B	$\geq$ 115.9 VAC
3.	Under-frequency	Bus A	$\geq$ 57 Hz
		Bus B	$\geq$ 57 Hz

3/4.6.4 CONTAINMENT AND DRYWELL ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The operability of the drywell isolation valves ensures that the drywell atmosphere will be directed to the suppression pool for the full spectrum of pipe breaks inside the drywell. Since the allowable value of drywell leakage is so large, individual drywell penetration leakage is not measured. By checking valve operability on any penetration which could contribute a large fraction of the design leakage, the total leakage is maintained at less than the design value.

All required Containment and Drywell Isolation Valves and their maximum isolation times are listed in the applicable plant procedures. The opening of locked or sealed closed (i.e., manual) containment isolation valves that results in an open penetration under administrative control includes the following considerations: (1) stationing an individual, who is in constant communication with the control room, at the valve controls, (2) instructing this individual to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

The maximum isolation times for containment and drywell automatic isolation valves are the times used in the FSAR accident analysis for valves with analytical closing times. For automatic isolation valves not having analytical closing times, closing times are derived by applying margins to previous valve closing test data obtained by using ASME Section XI criteria. Maximum closing times for these valves was determined by using a factor of two times the allowable (from previous test closure to next test closure) ASME Section XI margin and adding this to the previous test closure time.

3/4.6.5 DRYWELL VACUUM RELIEF

The safety-related functions of the four drywell vacuum relief subsystems are drywell isolation, proper operation of the drywell purge compressors, and OPERABILITY in a large-break LOCA to control weir wall overflow drag and impact loads. The drywell isolation and drywell purge OPERABILITY functions are discussed in Bases 3/4.6.4 and 3/4.6.7, respectively. Drywell vacuum relief is not required for hydrogen dilution or to protect drywell structural integrity in a design-basis accident.

DRYWELL VACUUM RELIEF (Continued)

To provide drywell vacuum relief, containment air is drawn through subsystems associated with three 10-inch lines penetrating the drywell. Two drywell post-LOCA vacuum relief subsystems are in a parallel arrangement connected to one of the three 10-inch vacuum relief lines penetrating the drywell. Each drywell post-LOCA vacuum relief subsystem consists of a motor-operated isolation valve in series with a check valve. OPERABILITY of either drywell post-LOCA vacuum relief subsystem assures OPERABILITY of the associated 10-inch vacuum relief line penetrating the drywell. Each of the two remaining 10-inch vacuum relief lines penetrating the drywell contains a drywell purge vacuum relief subsystem. Each drywell purge vacuum relief subsystem consists of a series arrangement of a motor-operated isolation valve and two check valves. Vacuum relief initiates at a differential pressure across the check valves of less than or equal to 1 psi.

Rapid weir wall overflow in a large-break LOCA could cause drag and impact loadings to essential equipment and systems in the drywell above the weir wall. Drywell negative pressure analysis for rapid weir wall overflow in a large-break LOCA assumes a vacuum breaker capability of $A/\sqrt{K} = 0.38\ ft^2$ thus requiring a minimum of two 10-inch drywell vacuum relief paths.

OPERABILITY requirements for the four drywell vacuum relief subsystems in relationship to continued plant operation are based on maintaining at least two of the three 10-inch drywell vacuum relief paths OPERABLE. However, to ensure that essential equipment is returned to service in a timely manner, continued plant operation is limited with only one 10-inch drywell vacuum relief line out of service. Plant operation is further limited when two of the three 10-inch lines are out of service to ensure prompt response to restore equipment to service or to place the plant in a condition where the equipment is not required. Plant operation is also limited with a drywell isolation vacuum breaker in the open position to help ensure that design drywell bypass leakage is not potentially exceeded. Position indication is required to be OPERABLE on all drywell vacuum breakers and motor-operated isolation valves to help identify potential drywell bypass leakage paths.

Surveillance requirements and intervals were chosen to reflect the importance associated with the drywell vacuum relief function and are based on good engineering judgement using previous accepted testing methods.

3/4.6.6 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The Auxiliary Building and Enclosure Building provide secondary containment during normal operation when the containment is sealed and in service. When the reactor is in COLD SHUTDOWN or REFUELING, the containment may be open and the Auxiliary Building and Enclosure Building then become the only containment.

The maximum isolation times for secondary containment automatic isolation dampers/valves are the times used in the FSAR accident analysis for dampers/valves with analytical closing times. For automatic isolation valves not having

SECONDARY CONTAINMENT (Continued)

analytical closing times, closing times are derived by applying margins to previous valve closing test data obtained by using ASME Section XI criteria. Maximum closing times for these valves was determined by using a factor of two times the allowable (from previous test closure to next test closure) ASME Section XI margin and adding this to the previous test closure time. All required Secondary Containment Isolation Valves/Dampers and their maximum isolation times are listed in applicable plant procedures.

Establishing and maintaining a vacuum in the Auxiliary Building and Enclosure Building with the standby gas treatment system once per 18 months, along with the surveillance of the doors, latches, dampers, valves, blind flanges, and rupture discs is adequate to ensure that there are no violations of the integrity of the secondary containment.

The OPERABILITY of the standby gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Continuous operation of the system with the heaters OPERABLE for 10 hours over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Primary containment electrical penetrations and penetration conductors are protected by either de-energizing circuits not required during reactor operation or demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers by periodic surveillance. A list of required circuit breakers and their required response times and trip setpoints is contained in the applicable plant procedures.

The surveillance requirements applicable to lower voltage circuit breakers provide assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

The OPERABILITY or bypassing of the motor operated valve thermal overload protection continuously or under accident conditions by integral bypass devices ensures that the thermal overload protection during accident conditions will not prevent safety-related valves from performing their function. The surveillance requirements for demonstrating the OPERABILITY or bypassing of the thermal overload protection continuously and or during accident conditions are in accordance with Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977. A list of required thermal overloads is contained in the applicable plant procedures.

The reactor protection system (RPS) electric power monitoring assemblies provide redundant protection to the RPS and other systems which receive power from the RPS buses by acting to disconnect the RPS from the power source circuits in the presence of an electrical fault in the power supply. The BASES for the functional requirements of the RPS are discussed in the BASES for Specification 3/4.3.1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 102 TO FACILITY OPERATING LICENSE NO. NPF-29

ENTERGY OPERATIONS, INC., ET AL.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By letter dated September 14, 1992, the licensee (Entergy Operations, Inc.) submitted a request for changes to the Grand Gulf Nuclear Station, Unit 1 (GGNS) Technical Specifications (TS). The proposed amendment would remove the TS tables that include lists of components referenced in individual specifications. In addition, the TS requirements have been modified to remove all references to these tables. Finally, the TS have been modified to state requirements in general terms that include the components listed in the tables removed from the TS. Guidance on the proposed TS changes was provided by Generic Letter 91-08, dated May 6, 1991.

2.0 EVALUATION

The licensee has proposed modifications of Table 3.3.2-1, "Isolation Actuation Instrumentation." The revisions proposed for TS Table 3.3.2-1 involve the deletion of information cross-referencing the isolation actuation instrumentation TS Table 3.6.4-1, which will be deleted. This requires removal of one column from the table, as well as deletion of several informational footnotes.

The licensee has proposed the removal of Table 3.6.4-1, "Containment and Drywell Isolation Valves," which is referenced in TS 3/4.6.4. With the removal of this table, the licensee has proposed to include the following statement of the limiting condition of operation (LCO) under TS 3.6.4:

> Each containment and drywell isolation valve shall be OPERABLE***.

The corresponding footnote will read:

> *** Normally closed or locked closed manual valves may be opened on an intermittent basis under administrative control.

In addition, the licensee has removed all references to Table 3.6.4-1 from the definitions of Drywell Integrity, Primary Containment Integrity, and Secondary Containment Integrity (TS 4.6.2.1, TS 4.6.1.1) and from the action requirements under TS 3.6.4 and TS 4.6.4.1 through 4.6.4.3. The definitions of Drywell Integrity and Primary Containment Integrity in TS 4.6.1.1, and TS 4.6.2.1 refer to TS 3.6.4 for an exception that is now covered by a footnote to the LCO rather than by the table removed from the TS. With the removal of the reference to Table 3.6.4-1, the licensee has proposed to state this exception as follows:

> ..., except for valves that are opened under administrative control as permitted by Specification 3.6.4.

The Surveillance Requirement for TS 4.6.4.1 is revised to state "Each 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 applicable isolation time" rather than stating the requirements in relation to the valves specified in Table 3.6.4-1. The Surveillance Requirement for TS 4.6.4.2 is revised to state "Each automatic 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." The Surveillance Requirement for TS 4.6.4.3 is revised to state "The isolation time of each power operated or automatic containment or drywell isolation valve## shall be determined to be within its limit when tested pursuant to Specification 4.0.5." The corresponding footnote will state:

> ## The provisions of Specification 4.0.4 are not applicable to automatic main steam line valves for entry into OPERATIONAL CONDITIONS 2 or 3, provided the surveillance is performed within 12 hours after reaching a reactor steam pressure of 600 psig and prior to entry into OPERATIONAL CONDITION 1.

With the removal of Table 3.6.4.1, the licensee has proposed to modify the footnote for TS 3.6.1.2 and TS 4.6.1.2 to state:

> # Except for those that are hydrostatically leak tested.

Similarly, the last footnote for TS 4.6.1.2 is modified to state:

> * Unless a hydrostatic test is required.

With the removal of the table of containment isolation valves, the operability requirements have been stated in general terms that apply to all containment isolation valves including those that are closed or locked closed. These valves are closed or locked closed consistent with the regulatory requirements for manually-operated valves that are used as containment isolation valves. Because opening these valves would be contrary to the operability requirements of these valves, the following footnote to the LCO has been proposed:

> Normally closed or locked closed manual valves may be opened on an intermittent basis under administrative control.

This change is consistent with the guidance in Generic Letter 91-08 and is, therefore, acceptable.

The licensee has proposed the removal of Table 3.6.6.2-1, "Secondary Containment Ventilation System Automatic Isolation Dampers/Valves," which is referenced in TS 3/4.6.6.2. With the removal of this table, the licensee has proposed to include the following statement of the LCO and Surveillance Requirements under TS 3/4.6.6.2:

> Each secondary containment ventilation system automatic isolation damper/valve shall be OPERABLE...

The NRC staff has concluded that the Secondary Containment Ventilation System Automation Isolation Dampers/Valves serve a purpose similar to the Containment Drywell Isolation Valves BWR-6 Mark III Containment and are therefore similarly acceptable.

The licensee has also proposed the removal of Table 3.8.4.1-1, "Primary Containment Penetration Conductor Overcurrent Protective Devices," which is referenced in TS 3/4.8.4.1. With the removal of this table, the licensee has proposed to include the following statement for the LCO under TS 3.8.4.1:

> Primary and backup containment penetration conductor overcurrent protective devices associated with each containment electrical penetration circuit shall be OPERABLE. The scope of these protective circuit breakers excludes those circuits for which credible fault currents would not exceed the electrical penetration design rating.

In addition, the licensee has proposed to revise TS 4.8.4.1 to remove references to Table 3.8.4.1-1. The surveillance requirement has been revised to state the following:

> Each of the primary and backup containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE:

The licensee has proposed the removal of Table 3.8.4.2-1, "Motor Operated Valves Thermal Overload Protection," which provides a list of valves with bypass devices that is referenced in TS 3.8.4.2. With the removal of this table, the licensee has proposed to include the following statement of the LCO under TS 3.8.4.2:

> The thermal overload protection of each valve whose motor operator performs a safety function shall be OPERABLE or shall be bypassed either continuously or only under accident conditions, as applicable, by an OPERABLE bypass device.

The licensee has proposed changes to the above TS that are consistent with the guidance provided in Generic Letter 91-08. In addition, the licensee has provided an updated copy of the Bases Section of TS 3.6.4 that addresses appropriate considerations for opening closed or locked closed valves on an intermittent basis. Finally, the licensee has confirmed that component lists removed from the TS have been updated to identify all components for which the TS requirements apply and are located in controlled plant procedures. These lists of components will also be included in the next revision of the Updated Final Safety Analysis Report.

On the basis of its review of this matter, the staff finds that the proposed changes to the TS for Grand Gulf Nuclear Station, Unit 1, are primarily administrative and do not alter the requirements set forth in the existing TS. However, the applicability of the operability requirements will extend to all containment isolation valves as noted in this evaluation. Overall, these changes will allow licensees to make corrections and updates to the list of components for which these TS requirements apply, under the provisions that control changes to plant procedures as specified in the Administrative Controls Section of the TS. Therefore, the staff finds that the proposed TS changes are acceptable.

3.0 STATE CONSULTATION

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

4.0 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 in surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (57 FR 47130). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

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

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