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. 97
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 June 26, 1991, as supplemented April 22, 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. 97, 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: May 20, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 97

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
3/4 1-5	3/4 1-5
3/4 3-9	3/4 3-9
---	3/4 3-9a
3/4 3-10	3/4 3-10
3/4 3-11	3/4 3-11
3/4 3-13	3/4 3-13
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3/4 3-15	3/4 3-15
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3/4 3-24	3/4 3-24
3/4 3-25	3/4 3-25
3/4 3-26	3/4 3-26
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30
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3/4 3-35	3/4 3-35
3/4 3-36	3/4 3-36
3/4 3-55	3/4 3-56
B 3/4 3-1	B 3/4 3-1
B 3/4 3-2	B 3/4 3-2
B 3/4 3-3	B 3/4 3-3
B 3/4 3-3a	B 3/4 3-3a
B 3/4 3-4	B 3/4 3-4

SURVEILLANCE REQUIREMENTS (Continued)

4.1.3.1.4 The scram discharge volume shall be determined OPERABLE by demonstrating:

a.* The scram discharge volume drain and vent valves OPERABLE, when control rods are scram tested from a normal control rod configuration of less than or equal to 50% ROD DENSITY at least once per 18 months, by verifying that the drain and vent valves:

1. Close within 30 seconds after receipt of a signal for control rods to scram, and

2. Open when the scram signal is reset.

b. Proper level sensor response by performance of a CHANNEL FUNCTIONAL TEST of the scram discharge volume scram and control rod block level instrumentation at least once per 92 days.

*The provisions of Specification 4.0.4 are not applicable provided this surveillance is performed at least once per 18 months.

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed the following limits:

Reactor Vessel Dome Pressure (psig)*	Maximum Insertion Times to Notch Position (Seconds)		
	43	29	13
950	0.31	[illegible]	[illegible]
1050	0.32	0.86	1.57

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the maximum scram insertion time of one or more control rods exceeding the maximum scram insertion time limits of Specification 3.1.3.2 as determined by Surveillance Requirement 4.1.3.2.a or b, operation may continue provided that:

1. For all "slow" control rods, i.e., those which exceed the limits of Specification 3.1.3.2, the individual scram insertion times do not exceed the following limits:

Reactor Vessel Dome Pressure (psig)*	Maximum Insertion Times to Notch Position (Seconds)		
	43	29	13
950	0.38	1.09	2.09
1050	0.39	1.14	2.22

2. For "fast" control rods, i.e., those which satisfy the limits of Specification 3.1.3.2, the average scram insertion times do not exceed the following limits:

Reactor Vessel Dome Pressure (psig)*	Maximum Average Insertion Times to Notch Position (Seconds)		
	43	29	13
950	0.30	0.78	1.40
1050	0.31	0.84	1.53

3. The sum of "fast" control rods with individual scram insertion times in excess of the limits of ACTION a.2 and of "slow" control rods does not exceed 7.

4. No "slow" control rod, "fast" control rod with individual scram insertion time in excess of the limits of ACTION a.2, or otherwise inoperable control rod occupy adjacent locations in any direction, including the diagonal, to another such control rod.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

*For in[illegible]iate reactor vessel dome pressure, the scram time criteria is determi[illegible] by linear interpolation at each notch position.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

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

b. With the number of OPERABLE channels less than required by the minimum OPERABLE channels per trip system requirement for one trip system:

1. If placing the inoperable channel(s) in the tripped condition would cause an isolation, the inoperable channel(s) shall be restored to OPERABLE status within 6 hours or the ACTION required by Table 3.3.2-1 for the affected trip function shall be taken.

OR

2. If placing the inoperable channel(s) in the tripped condition would not cause an isolation, the inoperable channel(s) and/or that trip system shall be placed in the tripped condition within

a) 12 hours for trip functions common to RPS instrumentation#; and

b) 24 hours for trip functions not common to RPS instrumentation#.

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

SURVEILLANCE REQUIREMENTS

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

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

4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation trip function shown in Table 3.3.2-3 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least once 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.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

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

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

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

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

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

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
2. MAIN STEAM LINE ISOLATION (Continued)				
f. Main Steam Line Tunnel Temperature - High	1	2	1, 2, 3	23
g. Main Steam Line Tunnel Δ Temp.- High	1	2	1, 2, 3	23
h. Manual Initiation	1, 10	2	1, 2, 3	22
3. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level-Low Low, Level 2	N.A.(c)(d)(h)	2	1, 2, 3, and #	25
b. Drywell Pressure - High***	N.A.(c)(d)(h)	2	1, 2, 3	25
c. Fuel Handling Area Ventilation Exhaust Radiation - High High	N.A.(j)	2	1, 2, 3, and *	25
d. Fuel Handling Area Pool Sweep Exhaust Radiation - High High	N.A.(j)	2	1, 2, 3, and *	25
e. Manual Initiation	N.A.(c)(d)(h) N.A.(c)(d)(h)	2 2	1, 2, 3 *	26 25
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	8	1	1, 2, 3	27
b. Δ Flow Timer	8	1	1, 2, 3	27
c. Equipment Area Temperature - High	8	1/room	1, 2, 3	27
d. Equipment Area Δ Temp. - High	8	1/room	1, 2, 3	27
e. Reactor Vessel Water Level - Low Low, Level 2	8	2	1, 2, 3	27

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
4. REACTOR WATER CLEANUP SYSTEM ISOLATION (Continued)				
f. Main Steam Line Tunnel Ambient Temperature - High	8	1	1, 2, 3	27
g. Main Steam Line Tunnel Δ Temp. - High	8	1	1, 2, 3	27
h. SLCS Initiation	8(i)	1	1, 2, 5##	30
i. Manual Initiation	8	2	1, 2, 3	26
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High				
1. Pressure	4	1	1, 2, 3	27
2. Time Delay	4	1	1, 2, 3	27
b. RCIC Steam Supply Pressure - Low	4, 9(m)	1	1, 2, 3	27
c. RCIC Turbine Exhaust Diaphragm Pressure - High	4	2	1, 2, 3	27
d. RCIC Equipment Room Ambient Temperature - High	4	1	1, 2, 3	27
e. RCIC Equipment Room Δ Temp. - High	4	1	1, 2, 3	27
f. Main Steam Line Tunnel Ambient Temperature - High	4	1	1, 2, 3	27
g. Main Steam Line Tunnel Δ Temp. - High	4	1	1, 2, 3	27
h. Main Steam Line Tunnel Temperature Timer	4	1	1, 2, 3	27

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
i. RHR Equipment Room Ambient Temperature - High	4	1/room	1, 2, 3	27
j. RHR Equipment Room Δ Temp. - High	4	1/room	1, 2, 3	27
k. RHR/RCIC Steam Line Flow - High	4	1	1, 2, 3	27
l. Manual Initiation	$4^{(k)}$	1	1, 2, 3	26
m. Drywell Pressure-High (ECCS-Division 1 and Division 2)	$9^{(m)}$	1	1, 2, 3	27
6. RHR SYSTEM ISOLATION				
a. RHR Equipment Room Ambient Temperature - High	3	1/room	1, 2, 3	28
b. RHR Equipment Room Δ Temp. - High	3	1/room	1, 2, 3	28
c. Reactor Vessel Water Level - Low, Level 3***	3 3(p)	2 2(p)	1, 2, 3 4, 5	28 31
d. Reactor Vessel (RHR Cut-in Permissive) Pressure - High***	$3^{(l)}$	2	1, 2, 3	28
e. Drywell Pressure - High***	$3^{(l)}$	2	1, 2, 3	28
f. Manual Initiation	3	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.

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

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

NOTES (Continued)

(b) 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.
(c) Also actuates the standby gas treatment system.
(d) Also actuates the control room emergency filtration system in the isolation mode of operation.
(e) Two upscale-Hi Hi, one upscale-Hi Hi and one downscale, or two downscale signals from the same trip system actuate the trip system and initiate isolation of the associated containment and drywell isolation valves.
(f) Also trips and isolates the mechanical vacuum pumps.
(g) Deleted.
(h) Also actuates secondary containment ventilation isolation dampers and valves per Table 3.6.6.2-1.
(i) Closes only RWCU system isolation valves G33-F001, G33-F004, and G33-F251.
(j) Actuates the Standby Gas Treatment System and isolates Auxiliary Building penetration of the ventilation systems within the Auxiliary Building.
(k) Closes only RCIC outboard valves. A concurrent RCIC initiation signal is required for isolation to occur.
(l) Valves E12-F037A and E12-F037B are closed by high drywell pressure. All other Group 3 valves are closed by high reactor pressure.
(m) Valve Group 9 requires concurrent drywell high pressure and RCIC Steam Supply Pressure-Low signals to isolate.
(n) Valves E12-F042A and E12-F042B are closed by Containment Spray System initiation signals.
(o) Also isolates valves E61-F009, E61-F010, E61-F056, and E61-F057 from Valve Group 7.
(p) 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 group shall be added to isolation time shown in Tables 3.6.4-1 and 3.6.6.2-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

###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	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**

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
2. MAIN STEAM LINE ISOLATION (Continued)				
f. Main Steam Line Tunnel Temperature - High	S	Q	A	1, 2, 3
g. Main Steam Line Tunnel Δ Temp. - High	S	Q	A	1, 2, 3
h. Manual Initiation	NA	Q(a)	NA	1, 2, 3
3. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2	S	Q	R(c)	1, 2, 3 and #
b. Drywell Pressure - High	S	Q	R(c)	1, 2, 3
c. Fuel Handling Area Ventilation Exhaust Radiation - High High	S	Q	A	1, 2, 3 and *
d. Fuel Handling Area Pool Sweep Exhaust Radiation - High High	S	Q	A	1, 2, 3 and *
e. Manual Initiation	NA	Q(a)	NA	1, 2, 3 and *
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	A	1, 2, 3
d. Equipment Area Ventilation Δ Temp. - High	S	Q	A	1, 2, 3
e. Reactor Vessel Water Level - Low Low, Level 2	S	Q	R(c)	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
4. REACTOR WATER CLEANUP SYSTEM ISOLATION (Continued)				
f. Main Steam Line Tunnel Ambient Temperature - High	S	Q	A	1, 2, 3
g. Main Steam Line Tunnel Δ Temp. - High	S	Q	A	1, 2, 3
h. SLCS Initiation	NA	$Q^{(b)}$	NA	1, 2, 5##
i. Manual Initiation	NA	$Q^{(a)}$	NA	1, 2, 3
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High				
1. Pressure	S	Q	$R^{(c)}$	1, 2, 3
2. Time Delay	NA	Q	Q	1, 2, 3
b. RCIC Steam Supply Pressure - Low	S	Q	$R^{(c)}$	1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure - High	S	Q	$R^{(c)}$	1, 2, 3
d. RCIC Equipment Room Ambient Temperature - High	S	Q	A	1, 2, 3
e. RCIC Equipment Room Δ Temp. - High	S	Q	A	1, 2, 3
f. Main Steam Line Tunnel Ambient Temperature - High	S	Q	A	1, 2, 3
g. Main Steam Line Tunnel Δ Temp. - High	S	Q	A	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION (Continued)				
h. Main Steam Line Tunnel Temperature Timer	NA	Q	Q	1, 2, 3
i. RHR Equipment Room Ambient Temperature - High	S	Q	A	1, 2, 3
j. RHR Equipment Room Δ Temp. - High	S	Q	A	1, 2, 3
k. RHR/RCIC Steam Line Flow - High	S	Q	R(c)	1, 2, 3
l. Manual Initiation	NA	Q(a)	NA	1, 2, 3
m. Drywell Pressure-High (ECCS Division 1 and Division 2)	S	Q	R(c)	1, 2, 3
6. RHR SYSTEM ISOLATION				
a. RHR Equipment Room Ambient Temperature - High	S	Q	A	1, 2, 3
b. RHR Equipment Room Δ Temp. - High	S	Q	A	1, 2, 3
c. Reactor Vessel Water Level - Low, Level 3	S	Q	R(c)	1, 2, 3, 4, 5
d. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	S	Q	R(c)	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
6. RHR SYSTEM ISOLATION (Continued)				
e. Drywell Pressure - High	S	Q	$R^{(c)}$	1, 2, 3
f. Manual initiation	NA	$Q^{(a)}$	NA	1, 2, 3

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

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

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

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

(a) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 92 days as part of circuitry required to be tested for automatic system isolation.

(b) Each train or logic channel shall be tested at least every other 92 days.

(c) Calibrate trip unit at least once per 92 days.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
C. DIVISION 3 TRIP SYSTEM			
1. HPCS SYSTEM			
a. Reactor Vessel Water Level - Low, Low, Level 2	4(b)	1, 2, 3, 4*, 5*	33
b. Drywell Pressure - High##	4(b)	1, 2, 3	33
c. Reactor Vessel Water Level-High, Level 8	2(c)	1, 2, 3, 4*, 5*	31
d. Condensate Storage Tank Level-Low	2(d)	1, 2, 3, 4*, 5*	34
e. Suppression Pool Water Level-High	2(d)	1, 2, 3, 4*, 5*	34
f. Manual Initiation##	1	1, 2, 3, 4*, 5*	32
D. LOSS OF POWER			
1. Division 1 and 2			
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	4	1, 2, 3, 4**, 5**	30
b. Deleted			
c. 4.16 kV Bus Undervoltage (Degraded Voltage)	4	1, 2, 3, 4**, 5**	30
2. Division 3			
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	4	1, 2, 3, 4**, 5**	30
b. 4.16 kV Bus Undervoltage (Degraded Voltage)	4	1, 2, 3, 4**, 5**	30

(a) A channel may be placed in an inoperable status for up to 6 hours during periods of required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.
(b) Also actuates the associated division diesel generator.
(c) Provides signal to close HPCS pump discharge valve only.
(d) Provides signal to HPCS pump suction valves only.
* Applicable when the system is required to be OPERABLE per Specification 3.5.2 or 3.5.3.
** Required when applicable ESF equipment is required to be OPERABLE.
\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 135 psig.
\## The injection function of Drywell Pressure - High and Manual Initiation are not required to be OPERABLE with indicated reactor vessel water level on the wide range instrument greater than Level 8 setpoint coincident with the reactor pressure less than 600 psig.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. With one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or declare the associated system(s) inoperable.

b. With more than one channel inoperable, declare the associated system(s) inoperable.

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, within 24 hours declare the associated ADS trip system or ECCS inoperable.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 24 hours or declare the associated ADS trip system or ECCS inoperable.

ACTION 33 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel(s) in the tripped condition within 24 hours or declare the HPCS system inoperable.

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within 24 hours or declare the HPCS system inoperable.

ACTION 35 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel(s) in the tripped condition within 24 hours or declare the associated system(s) inoperable.

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. LPCI Pump A Start Time Delay Relay	NA	Q	Q	1, 2, 3, 4*, 5*
d. Manual Initiation	NA	$R^{(b)}$	NA	1, 2, 3, 4*, 5*
e. Reactor Vessel Pressure - Low (Injection Permissive)	S	Q	$R^{(a)}$	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 Initiation 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. Manual Initiation	NA	$R^{(b)}$	NA	1, 2, 3
h. ADS Bypass Timer (High Drywell Pressure)	NA	Q	Q	1, 2, 3
i. Manual Inhibit	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 2 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. LPCI Pump B Start Time Delay Relay	NA	Q	Q	1, 2, 3, 4*, 5*
d. Manual Initiation	NA	R$^{(b)}$	NA	1, 2, 3, 4*, 5*
e. Reactor Vessel Pressure - Low (Injection Permissive)	S	Q	R$^{(a)}$	1, 2, 3, 4*, 5*

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
B. DIVISION 2 TRIP SYSTEM (Continued)				
2. AUTOMATIC DEPRESSURIZATION SYSTEM				
TRIP SYSTEM "B"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	$R^{(a)}$	1, 2, 3
b. Drywell Pressure-High	S	Q	$R^{(a)}$	1, 2, 3
c. ADS Initiation 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. Manual Initiation	NA	$R^{(b)}$	NA	1, 2, 3
g. ADS Bypass Timer (High Drywell Pressure)	NA	Q	Q	1, 2, 3
h. Manual Inhibit	NA	R	NA	1, 2, 3
C. DIVISION 3 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. Manual Initiation##	NA	$R^{(b)}$	NA	1, 2, 3, 4*, 5*

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
D. LOSS OF POWER				
1. Division 1 and 2				
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	NA	$M^{(e)}$	R	1, 2, 3, 4**, 5**
b. Deleted				
c. 4.16 kV Bus Undervoltage (Degraded Voltage)	NA	$M^{(e)}$	R	1, 2, 3, 4**, 5**
2. Division 3				
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	NA	NA	R	1, 2, 3, 4**, 5**
b. 4.16 kV Bus Undervoltage (Degraded Voltage)	NA	NA	R	1, 2, 3, 4**, 5**

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

NOTATION

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 135 psig.

\## The injection function of Drywell Pressure - High and Manual Initiation are not required to be OPERABLE with indicated reactor vessel water level on the wide range instrument greater than Level 8 setpoint coincident with the reactor pressure less than 600 psig.

* Applicable when the system is required to be OPERABLE per Specification 3.5.2 or 3.5.3.

** Required when ESF equipment is required to be OPERABLE.

(a) Calibrate trip unit at least once per 92 days.

(b) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 92 days as a part of circuitry required to be tested for automatic system actuation.

(c) DELETED

(d) DELETED

(e) Functional Testing of Time Delay Not Required

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.
b. Preserve the integrity of the reactor coolant system.
c. Minimize the energy which must be absorbed following a loss-of-coolant accident, and
d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P, "Technical Specification Improvement Analyses for BWR Reactor Protection System," as approved by the NRC and documented in the SER (letter T. A. Pickens from A. Thadani dated July 15, 1987). The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the accident analysis. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance. Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. Negative barometric pressure fluctuations are accounted for in the trip setpoints and allowable values specified for drywell pressure-high. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

ISOLATION ACTUATION INSTRUMENTATION (Continued)

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with: (1) NEDC-30851P-A, Supplement 2, "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation" as approved by the NRC and documented in the NRC Safety Evaluation Report (letter to D. N. Grace from C. E. Rossi dated January 6, 1989) and (2) NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation" as approved by the NRC and documented in the NRC Safety Evaluation Report (letter to S. D. Floyd from C. E. Rossi dated June 18, 1990).

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D.C. operated valves, a 3 second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 10 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 10 second diesel startup. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 10 second delay. It follows that checking the valve speeds and the 10 second time for emergency power establishment will establish the response time for the isolation functions. However, to enhance overall system reliability and to monitor instrument channel response time trends, the isolation actuation instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Negative barometric pressure fluctuations are accounted for in the trip setpoints and allowable values specified for drywell pressure-high. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30936P-A, Parts 1 and 2, "Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation)" as approved by the NRC and documented in the NRC Safety Evaluation Reports (letter to D. N. Grace from A. C. Thadani dated December 9, 1988 (Part 1) and letter to D. N. Grace from C. E. Rossi dated December 9, 1988 (Part 2)).

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION (Continued)

Operation with a trip set less conservative than its Trip Setpoint but w[illegible] specified Allowable Value is acceptable on the basis that the [illegible] between each Trip Setpoint and the Allowable Value i[illegible] equal to or g[illegible] the drift allowance assumed for each trip in the safety analyses.

3/[illegible] [illegible]CULATION PUMP TRIP ACTUATION INSTRUMENTATION

Th[illegible] [illegible]cipated transient without scram recirculation pump trip (ATWS-RPT) system [illegible]des a means of limiting the consequences of the unlikely occurrence of a [illegible] to scram during an anticipated transient. The response of the plant [illegible] postulated event has been evaluated in General Electric Company report NED[illegible] 12408 dated March 1987. The results of the analysis show that the Grand Gulf ATWS-RPT design provides adequate protection for these events in which the normal scram paths fail.

The ATWS-RPT provides fully redundant trip of the recirculation pump motors so that the pumps coast down to zero speed. This trip function reduces core flow creating steam voids in the core, thereby decreasing power generation and limiting any power or pressure excursions. The Grand Gulf ATWS-RPT design provides compliance with the requirements of the NRC ATWS Rule 1[illegible]CFR50.62.

The ATWS-RPT and Alternate Rod Insertion (ARI) system use common setpoints and trip channels ([illegible]transmitters and trip systems). Therefore, the ARI trip function [illegible] the RPT trip function will be initiated simultaneously. The instrumentation setpoints for the RPV pressure and water level trip channels are established such that the normal scram paths for these variables would already be initiated.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during [illegible]o of the most limiting pressurization eve[illegible]s. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sen[illegible]r from each of two turbine control valves provides input to the EOC-RPT system; fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a closure sensor for each of two turbine stop valves provides input to one EOC-RPT system; a closure sensor from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION (Continued)

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 40% of RATED THERMAL POWER are annunciated in the control room. The automatic bypass setpoint is feedwater temperature dependent due to the subcooling changes that affect the turbine first-stage pressure-reactor power relationship. For RATED THERMAL POWER operation with feedwater temperature greater than or equal to 420°F, an allowable setpoint of $\leq$ 26.9% of control valve wide open turbine first-stage pressure is provided for the bypass function. This setpoint is also applicable to operation at less than RATED THERMAL POWER with the correspondingly lower feedwater temperature. The allowable setpoint is reduced to $\leq$ 22.5% of control valve wide open turbine first-stage pressure for RATED THERMAL POWER operation with a feedwater temperature between 370°F and 420°F. Similarly, the reduced setpoint is applicable to operation at less than RATED THERMAL POWER with the corresponding lower feedwater temperature.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 190 ms. Included in this time are: the response time of the sensor, the response time of the system logic and the breaker interruption time. Breaker interruption time includes both breaker response time and the manufacturer's design arc suppression time of 12 ms.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater then the drift allowance assumed for each trip in the safety analyses.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits. The trip logic is arranged so that a trip in any one of the inputs will results in a control rod block.

The OPERABILITY of the control rod block instrumentation in OPERATIONAL CONDITION 5 is to provide diversity of rod block protection to the one-rod-out interlock.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION (Continued)

Specified surveillance intervals have been determined in accordance with NEDC-30851P-A, Supplement 1, "Technical Specification Improvement Analysis f BWR Control Rod Block Instrumentation" as approved by the NRC and documented in the NRC Safety Evaluation Report (letter to D. N. Grace from C. E. Rossi dated September 22, 1988).

Operation with a trip set less conservative than its p Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that: (1) the radiation levels are continually measured in the areas served by the individual channels; (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of NUREG-0737, "Clarification of TMI Action Plan Requirements," November, 1980.

3/4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit.

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February, 1972.

3/4.3.7.4 REMOTE SHUTDOWN SYSTEM INSTRUMENTATION AND CONTROLS

The OPERABILITY of the remote shutdown system instrumentation and controls ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR 50.