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Mr. Donald C.Shelton
Vice President Nuclear - Perry
Centerior Service Company
P. O. Box 97, A200
Perry, OH 44081

SUBJECT: AMENDMENT NO. 75 TO FACILITY OPERATING LICENSE NO. NPF-58 - PERRY NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NOS. M91880 and M92522)

Dear Mr. Shelton:

The Commission has issued the enclosed Amendment No. 75 to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant, Unit No. 1. This amendment revises the Technical Specifications in response to your applications dated March 24, June 9, and June 30, 1995.

This amendment revises the Technical Specifications to allow a one-time extension for the performance of certain Surveillance Requirements (SRs). Affected SRs include penetration leak rate testing, valve operability testing, instrument calibration, response time testing, and logic system functional tests.

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

Sincerely,
Original signed by:
Jon B. Hopkins, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-440

Enclosures: 1. Amendment No. 75 to License No. NPF-58
2. Safety Evaluation

cc w/encls: See next page

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

November 29, 1995

Mr. Donald C. Shelton
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Centerior Service Company
P. O. Box 97, A200
Perry, OH 44081

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cc w/encls: See next page

Mr. Donald C. Shelton
Centerior Service Company

Perry Nuclear Power Plant
Unit Nos. 1 and 2

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

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.

DOCKET NO. 50-440

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 75
License No. NPF-58

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

   A. The application for amendment by The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, and Toledo Edison Company (the licensees) dated March 24, June 9, and June 30, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, 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-58 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. 75 are hereby incorporated into this license. The Cleveland Electric Illuminating Company 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 and shall be implemented not later than 90 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jon B. Hopkins

Jon B. Hopkins, Senior Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: November 29, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 75

FACILITY OPERATING LICENSE NO. NPF-58

DOCKET NO. 50-440

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.

Remove	Insert
3/4 0-2	3/4 0-2
3/4 3-1	3/4 3-1
--	3/4 3-1a
3/4 3-6	3/4 3-6
3/4 3-7	3/4 3-7
3/4 3-8a	3/4 3-8a
3/4 3-10	3/4 3-10
3/4 3-21	3/4 3-21
3/4 3-22	3/4 3-22
3/4 3-23	3/4 3-23
3/4 3-24	3/4 3-24
3/4 3-25	3/4 3-25
3/4 3-26	3/4 3-26
3/4 3-27	3/4 3-27
3/4 3-35	3/4 3-35
3/4 3-36	3/4 3-36
3/4 3-37	3/4 3-37
3/4 3-38	3/4 3-38
3/4 3-39	3/4 3-39
3/4 3-50	3/4 3-50
3/4 3-54	3/4 3-54
3/4 3-55	3/4 3-55
3/4 3-59	3/4 3-59
3/4 3-60	3/4 3-60
3/4 3-73	3/4 3-73
3/4 3-76	3/4 3-76
3/4 3-77	3/4 3-77
3/4 3-80	3/4 3-80
3/4 3-98	3/4 3-98
3/4 3-102	3/4 3-102
3/4 4-7	3/4 4-7
--	3/4 4-7a
3/4 4-8	3/4 4-8
3/4 4-9	3/4 4-9
3/4 4-11	3/4 4-11
3/4 5-4	3/4 5-4
3/4 5-5	3/4 5-5
3/4 6-5	3/4 6-5
3/4 6-29	3/4 6-29

Remove	Insert
3/4 6-47	3/4 6-47
3/4 7-1	3/4 7-1
3/4 7-2	3/4 7-2
3/4 7-4	3/4 7-4
3/4 7-8	3/4 7-8
3/4 7-10	3/4 7-10
3/4 8-4	3/4 8-4
3/4 8-5	3/4 8-5

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL CONDITIONS as required to comply with ACTION requirements.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable* as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

* 2-year Surveillance Intervals may be extended to the completion of the fifth refueling outage.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

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

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

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

2. Within 6 hours, place the inoperable channel(s) in one trip system and/or that trip system** in the tripped condition,* and

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

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

SURVEILLANCE REQUIREMENTS

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

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

---

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

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

\# Channel Calibration period may be extended as identified by note 'n' on Table 4.3.1.1-1.

\## Logic System Functional Test period may be extended as identified by note 'o' on Table 4.3.1.1-1.

SURVEILLANCE REQUIREMENTS (continued)

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

4.3.1.4 The provisions of Specification 4.0.4 are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for the Intermediate Range Monitors for entry into their applicable OPERATIONAL CONDITIONS (as shown in Table 4.3.1.1-1) from OPERATIONAL CONDITION 1, provided the surveillances are performed within 12 hours after such entry.

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

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

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

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

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

\# Measured from start of turbine control valve fast closure.

\## Response time testing may be extended to be performed during the fifth refueling outage.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reaching 25% of RATED THERMAL POWER. To functionally implement this protective function during entry into single loop operation, APRM channel gain adjustments may be made in lieu of adjusting the APRM Flow Biased Simulated Thermal Power-High Trip Setpoint and Allowable Value equations for a period not to exceed 72 hours, provided the criteria in Note b to Table 2.2.1-1 are met. Any APRM channel gain adjustments made in compliance with Specifications 2.2.1 and 3.3.1 shall not be included in determining the absolute difference.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 1000 MWD/T using the TIP system.

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

(h) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing loop flow (APRM % flow).

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

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

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

(l) This function is not required to be OPERABLE when Drywell Integrity is not required.

(m) The CHANNEL CALIBRATION shall exclude the flow reference transmitters, these transmitters shall be calibrated at least once per 18 months, except that this test may be extended to be performed during the fifth refueling outage.

(n) CHANNEL CALIBRATION may be extended to be performed during the fifth refueling outage.

(o) LOGIC SYSTEM FUNCTIONAL TESTING may be extended to be performed during the fifth refueling outage.

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.

CHANNEL CALIBRATION period may be extended as identified by note 'c' on Table 4.3.2-1.

LOGIC SYSTEM FUNCTIONAL TEST period may be extended as identified by note 'd' on Table 4.3.2-1.

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

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low, Level 2	NA
b. Drywell Pressure - High	NA
c. Containment and Drywell Purge Exhaust Plenum Radiation - High[(b)]	$\leq 10^{(a),(c)}$
d. Reactor Vessel Water Level - Low, Level 1	NA
e. Manual Initiation	NA
2. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level - Low, Level 1	≤ 1.0*/$\leq 10^{(a)**(c)}$
b. Main Steam Line Radiation - High	NA
c. Main Steam Line Pressure - Low	≤ 1.0*/$\leq 10^{(a)**(c)}$
d. Main Steam Line Flow - High	≤ 0.5*/$\leq 10^{(a)**(c)}$
e. Condenser Vacuum - Low	NA
f. Main Steam Line Tunnel Temperature - High	NA
g. Main Steam Line Tunnel Δ Temperature-High	NA
h. Turbine Building Main Steam Line Temperature - High	NA
i. Manual Initiation	NA
3. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low, Level 2	NA
b. Drywell Pressure - High	NA
c. Manual Initiation	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. Δ Flow - High	NA
b. Δ Flow Timer	NA
c. Equipment Area Temperature - High	NA
d. Equipment Area Δ Temperature - High	NA
e. Reactor Vessel Water Level - Low, Level 2	NA
f. Main Steam Line Tunnel Ambient Temperature - High	NA
g. Main Steam Line Tunnel Δ Temperature-High	NA
h. SLCS Initiation	NA
i. Manual Initiation	NA

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	NA
b. RCIC Steam Supply Pressure - Low	NA
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
d. RCIC Equipment Room Ambient Temperature - High	NA
e. Deleted	
f. Main Steam Line Tunnel Ambient Temperature - High	NA
g. Main Steam Line Tunnel Δ Temperature - High	NA
h. Main Steam Line Tunnel Temperature Timer	NA
i. RHR Equipment Room Ambient Temperature - High	NA
j. RHR Equipment Room Δ Temperature - High	NA
k. RCIC Steam Line Flow High Timer	NA
l. Drywell Pressure - High	NA
m. Manual Initiation	NA
6. RHR SYSTEM ISOLATION	
a. RHR Equipment Area Ambient Temperature - High	NA
b. RHR Equipment Area Δ Temperature - High	NA
c. RHR/RCIC Steam Line Flow - High	NA
d. Reactor Vessel Water Level - Low, Level 3	NA
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA
f. Drywell Pressure - High	NA
g. Manual Initiation	NA

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

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

(c) Response time testing may be extended to be performed during the fifth refueling outage.

* Isolation system instrumentation response time for MSIV's only. No diesel generator delays assumed.

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

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

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, Level 2	S	Q	R(b)(c)(d)	1, 2, 3 and #
b. Drywell Pressure - High ##	S	Q	R(b)	1, 2, 3
c. Containment and Drywell Purge Exhaust Plenum Radiation - High	S	Q	R(d)	1, 2, 3 and *
d. Reactor Vessel Water Level - Low, Level 1	S	Q	R(b)(c)(d)	1, 2, 3 and #
e. Manual Initiation	NA	R(d)	NA	1, 2, 3 and *
2. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 1	S	Q	R(b)(c)(d)	1, 2, 3
b. Main Steam Line Radiation - High	S	Q	R(c)(d)	***
c. Main Steam Line Pressure - Low	S	Q	R(b)	1
d. Main Steam Line Flow - High	S	Q	R(b)	1, 2, 3
e. Condenser Vacuum - Low	S	Q	R(b)	1, 2**, 3**
f. Main Steam Line Tunnel Temperature - High	S	Q	R	1, 2, 3
g. Main Steam Line Tunnel Δ Temperature - High	S	Q	R	1, 2, 3
h. Turbine Building Main Steam Line Temperature - High	S	Q	R	1, 2, 3
i. Manual Initiation	NA	R	NA	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
3. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low, Level 2	S	Q	$R^{(b)(c)(d)}$	1, 2, 3 and #
b. Drywell Pressure - High ##	S	Q	$R^{(b)}$	1, 2, 3
c. Manual Initiation	NA	R	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	R	1, 2, 3
c. Equipment Area Temperature - High	S	Q	R	1, 2, 3
d. Equipment Area Ventilation Δ Temperature - High	S	Q	R	1, 2, 3
e. Reactor Vessel Water Level - Low, Level 2	S	Q	$R^{(b)(c)(d)}$	1, 2, 3
f. Main Steam Line Tunnel Ambient Temperature - High	S	Q	R	1, 2, 3
g. Main Steam Line Tunnel Δ Temperature - High	S	Q	R	1, 2, 3
h. SLCS Initiation	NA	$Q^{(a)}$	NA	1, 2, 3
i. Manual Initiation	NA	R	NA	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	S	Q	R(b)	1, 2, 3
b. RCIC Steam Supply Pressure - Low	S	Q	R(b)	1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure - High	S	Q	R(b)	1, 2, 3
d. RCIC Equipment Room Ambient Temperature - High	S	Q	R	1, 2, 3
e. Deleted				
f. Main Steam Line Tunnel Ambient Temperature - High	S	Q	R	1, 2, 3
g. Main Steam Line Tunnel Δ Temperature - High	S	Q	R	1, 2, 3
h. Main Steam Line Tunnel Temperature Timer	NA	Q	R	1, 2, 3
i. RHR Equipment Room Ambient Temperature - High	S	Q	R	1, 2, 3
j. RHR Equipment Room Δ Temperature - High	S	Q	R	1, 2, 3
k. RCIC Steam Line Flow High Timer	NA	Q	R	1, 2, 3
l. Drywell Pressure - High	S	Q	R(b)	1, 2, 3
m. Manual Initiation	NA	R(d)	NA	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
6. RHR SYSTEM ISOLATION				
a. RHR Equipment Area Ambient Temperature - High	S	Q	R	1, 2, 3
b. RHR Equipment Area Δ Temperature - High	S	Q	R	1, 2, 3
c. RHR/RCIC Steam Line Flow - High	S	Q	R(b)	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3 ##	S	Q	R(b)(c)(d)	1, 2, 3
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	S	Q	R(b)(c)(d)	1, 2, 3
f. Drywell Pressure - High ##	S	Q	R(b)(d)	1, 2, 3
g. Manual Initiation	NA	R(d)	NA	1, 2, 3

* When handling irradiated fuel in the primary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.
** When any turbine stop valve is greater than 90% open and/or the key locked bypass switch is in the normal position.
*** OPERATIONAL CONDITION 1 or 2 when the mechanical vacuum pump lines are not isolated.
\# During CORE ALTERATION and operations with a potential for draining the reactor vessel.
(a) Each train or logic channel shall be tested at least every other 92 days.
(b) Calibrate trip unit setpoint at least once per 92 days.
\## These Trip Functions (1b, 3b, 6d, and 6f) utilize instruments which are common to RPS instrumentation.
(c) CHANNEL CALIBRATION may be extended to be performed during the fifth refueling outage.
(d) LOGIC SYSTEM FUNCTIONAL TEST may be extended to be performed during the fifth refueling outage.

3.4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function shown in Table 3.3.3-3 shall be demonstrated to be within the 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 ECCS trip system.

CHANNEL CALIBRATION testing period may be extended as identified by note 'b' on Table 4.3.3.1-1.
LOGIC SYSTEM FUNCTIONAL TEST period may be extended as identified by note 'c' on Table 4.3.3.1-1.
RESPONSE TIME test period may be extended as identified by note 'a' on Table 3.3.3-3.

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

ECCS	RESPONSE TIME (Seconds)
A. DIVISION 1 TRIP SYSTEM	
1. RHR-A (LPCI MODE) AND LPCS SYSTEM	
a. Reactor Vessel Water Level - Low, Level 1	≤ 37[a]
b. Drywell Pressure - High	≤ 37
c. LPCS Pump Discharge Flow - Low (Bypass)	NA
d. Reactor Vessel Pressure - Low (LPCS Injection Valve Permissive)	NA
e. Reactor Vessel Pressure - Low (LPCI Injection Valve Permissive)	NA
f. LPCI Pump A Start Time Delay Relay	NA
g. LPCI Pump A Discharge Flow - Low (Bypass)	NA
h. Manual Initiation	NA
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"	
a. Reactor Vessel Water Level - Low, Level 1	NA[a]
b. Manual Inhibit	NA
c. ADS Timer	NA
d. Reactor Vessel Water Level - Low, Level 3 (Permissive)	NA
e. LPCS Pump Discharge Pressure - High (Permissive)	NA
f. LPCI Pump A Discharge Pressure - High (Permissive)	NA
g. Manual Initiation	NA
B. DIVISION 2 TRIP SYSTEM	
1. RHR B AND C (LPCI MODE)	
a. Reactor Vessel Water Level - Low, Level 1	≤ 37[a]
b. Drywell Pressure - High	≤ 37
c. Reactor Vessel Pressure - Low (LPCI Injection Valve Permissive)	NA
d. LPCI Pump B Start Time Delay Relay	NA
e. LPCI Pump Discharge Flow - Low (Bypass)	NA
f. Manual Initiation	NA

TABLE 3.3.3-3 (Continued)

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

TRIP FUNCTION	RESPONSE TIME (Seconds)
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"	
a. Reactor Vessel Water Level - Low, Level 1	NA
b. Manual Inhibit	NA
c. ADS Timer	NA
d. Reactor Vessel Water Level - Low, Level 3 (Permissive)	NA
e. LPCI Pump B and C Discharge Pressure - High (Permissive)	NA
f. Manual Initiation	NA
C. DIVISION 3 TRIP SYSTEM	
1. HPCS SYSTEM	
a. Reactor Vessel Water Level - Low, Level 2	≤ 27[(a)]
b. Drywell Pressure - High	≤ 27[(a)]
c. Reactor Vessel Water Level - High, Level 8	NA
d. Condensate Storage Tank Level - Low	NA
e. Suppression Pool Water Level - High	NA
f. HPCS Pump Discharge Pressure - High	NA
g. HPCS System Flow Rate - Low	NA
h. Manual Initiation	NA
D. LOSS OF POWER	
1. 4.16 kv Emergency Bus Undervoltage# (Loss of Voltage)	NA
2. 4.16 kv Emergency Bus Undervoltage# (Degraded Voltage)	NA

\# The Loss of Voltage and Degraded Voltage functions are common to Division 1, Division 2, and Division 3.

(a) May be extended to the completion of the fifth refueling outage.

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

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
C. DIVISION 3 TRIP SYSTEM				
1. HPCS SYSTEM				
a. Reactor Vessel Water Level - Low, Level 2	S	Q	$R^{(a)(b)(c)}$	1, 2, 3, 4*, 5*
b. Drywell Pressure-High ##	S	Q	$R^{(a)(c)}$	1, 2, 3
c. Reactor Vessel Water Level - High, Level 8	S	Q	$R^{(a)(b)(c)}$	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. HPCS Pump Discharge Pressure - High	S	Q	$R^{(a)}$	1, 2, 3, 4*, 5*
g. HPCS System Flow Rate - Low	S	Q	$R^{(a)}$	1, 2, 3, 4*, 5*
h. Manual Initiation##	NA	$R^{(c)}$	NA	1, 2, 3, 4*, 5*
D. LOSS OF POWER				
1. 4.16 kV Emergency Bus Under-voltage (Loss of Voltage)	NA	NA	$R^{(b)(c)}$	1, 2, 3, 4**, 5**
2. 4.16 kv Emergency Bus Under-voltage (Degraded Voltage)	S	M	$R^{(b)(c)}$	1, 2, 3, 4**, 5**

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 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 the Level 8 setpoint coincident with reactor pressure less than 450 psig.

* 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 setpoint at least once per 92 days.

(b) CHANNEL CALIBRATION may be extended to the completion of the fifth refueling outage.

(c) LOGIC SYSTEM FUNCTIONAL TEST may be extended to the completion of the fifth refueling outage.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5 The reactor core isolation cooling (RCIC) system actuation instrumentation channels shown in Table 3.3.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.5-2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

a. With a RCIC system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.5-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more RCIC system actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.5-1.

SURVEILLANCE REQUIREMENTS

4.3.5.1 Each RCIC system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.5.1-1.#

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

Channel Calibration period may be extended as identified by note 'b' on Table 4.3.5.1-1.

LOGIC SYSTEM FUNCTIONAL TEST period may be extended as identified by note 'c' on Table 4.3.5.1-1.

TABLE 4.3.5.1-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level - Low, Level 2	S	Q	R(a)(b)(c)
b. Reactor Vessel Water Level - High, Level 8	S	Q	R(a)(b)(c)
c. Condensate Storage Tank Level - Low	S	Q	R(a)
d. Suppression Pool Water Level - High	S	Q	R(a)
e. Manual Initiation	NA	R	NA

(a) Calibrate trip unit setpoint at least once per 92 days.
(b) CHANNEL CALIBRATION may be extended to be performed during the fifth refueling outage.
(c) LOGIC SYSTEM FUNCTIONAL TEST may be extended to be performed during the fifth refueling outage.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

4.3.6.2 The provisions of Specification 4.0.4 are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for the Intermediate Range Monitors and Source Range Monitors for entry into their applicable OPERATIONAL CONDITIONS (as shown in Table 4.3.6-1) from OPERATIONAL CONDITION 1 provided the surveillances are performed within 12 hours after such entry.

* The APRM flow biased instrumentation need not be declared inoperable upon entering single recirculation loop operation provided the setpoints are adjusted within 8 hours per Specification 3.4.1.1.

Channel Calibration may be extended by note 'c' on Table 4.3.6-1. Channel Functional Test period may be extended as identified by note 'e' on Table 4.3.6-1.

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. ROD PATTERN CONTROL SYSTEM				
a. Low Power Setpoint	NA	S/U(b), Q	SA#	1, 2
b. RWL - High Power Setpoint	NA	S/U(b), Q	SA#	1
2. APRM				
a. Flow Biased Neutron Flux - Upscale				
1) Flow Biased	NA	S/U(b), Q	SA(c)##	1
2) High Flow Clamped	NA	S/U(b), Q	SA(c)	1
b. Inoperative	NA	S/U(b), Q	NA	1, 2, 5
c. Downscale	NA	S/U(b), Q	SA	1
d. Neutron Flux - Upscale, Startup	NA	S/U(b), Q	SA	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U(b),W	NA	2**, 5
b. Upscale	NA	S/U(b),W(d)	R	2**, 5
c. Inoperative	NA	S/U(b),W	NA	2**, 5
d. Downscale	NA	S/U(b),W(d)	R	2**, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U(b),W	NA	2, 5
b. Upscale	NA	S/U(b),W(d)	R	2, 5
c. Inoperative	NA	S/U(b),W	NA	2, 5
d. Downscale	NA	S/U(b),W(d)	R	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	S/U(b),Q	SA(c)##	1
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	R(e)	NA	3, 4

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

b. Within 7 days prior to startup.

c. The CHANNEL CALIBRATION shall exclude the flow reference transmitters, these transmitters shall be calibrated at least once per 18 months, except that this test may be extended to the completion of the fifth refueling outage as noted by ##.

d. Trip setpoints are verified during weekly CHANNEL FUNCTIONAL TESTS.

e. May be extended to the completion of the fifth refueling outage.

---

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

** With IRMs on range 2 or below.

\# Calibrate trip unit setpoint at least once per 92 days.

INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION AND CONTROLS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

b. With the number of OPERABLE remote shutdown system controls less than required in Table 3.3.7.4-1, restore the inoperable control(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

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

TABLE 4.3.7.4-1

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Vessel Pressure	M	R[a][b]
2. Reactor Vessel Water Level	M	R[a][b]
3. Safety/Relief Valve Position	M	NA
4. Suppression Pool Water Level	M	R
5. Suppression Pool Water Temperature	M	R
6. Drywell Pressure	M	R
7. Drywell Temperature	M	R
8. RHR System Flow	M	R
9. Emergency Service Water Flow to RHR Heat Exchanger	M	R
10. Emergency Service Water Flow to Emergency Closed Cooling Heat Exchanger	M	R
11. RCIC System Flow	M	R
12. RCIC Turbine Speed	M	R
13. Emergency Closed Cooling System Flow	M	R
14. Inboard MSIV Position	M	NA

(a) CHANNEL CALIBRATION may be extended to be performed during the fifth refueling outage.

(b) Operability testing may be extended to be performed during the fifth refueling outage.

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.7.5-1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

TABLE 4.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	APPLICABLE OPERATIONAL CONDITIONS
1. Reactor Vessel Pressure	M	R	1, 2, 3
2. Reactor Vessel Water Level			
a. Fuel Zone	M	R(a)	1, 2, 3
b. Wide Range	M	R(a)	1, 2, 3
3. Suppression Pool Water Level	M	R	1, 2, 3
4. Suppression Pool Water Temperature	M	R	1, 2, 3
5. Primary Containment Pressure	M	R	1, 2, 3
6. Primary Containment Air Temperature	M	R	1, 2, 3
7. Drywell Pressure	M	R	1, 2, 3
8. Drywell Air Temperature	M	R	1, 2, 3
9. Primary Containment and Drywell Hydrogen Concentration Analyzer and Monitor	NA	Q*	1, 2, 3
10. Safety/Relief Valve Position Indicators	M	R	1, 2, 3
11. Primary Containment/Drywell Area Gross Gamma Radiation Monitors	M	R**(a)	1, 2, 3
12. Offgas Ventilation Exhaust Monitor#	M	R	1, 2, 3
13. Turbine Building/Heater Bay Ventilation Exhaust Monitor#	M	R	1, 2, 3
14. Unit 1 Vent Monitor#	M	R	1, 2, 3
15. Unit 2 Vent Monitor#	M	R	1, 2, 3
16. Neutron Flux			
a. Average Power Range	M	R	1, 2, 3
b. Intermediate Range	M	R	1, 2, 3
c. Source Range	M	R	1, 2, 3
17. Primary Containment Isolation Valve Position	M	R(a)	1, 2, 3

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

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

\# High and intermediate range D19 system noble gas monitors.

(a) CHANNEL CALIBRATION may be extended to be performed during the fifth refueling outage.

3/4.3.9 PLANT SYSTEMS ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.9-1.

ACTION:

a. With a plant system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.9-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

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

SURVEILLANCE REQUIREMENTS

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

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

CHANNEL CALIBRATION period may be extended as identified by Note 'a' on Table 4.3.9.1-1.

LOGIC SYSTEM FUNCTIONAL TEST period may be extended by Note 'b' on Table 4.3.9.1-1.

TABLE 4.3.9.1-1

PLANT SYSTEMS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. CONTAINMENT SPRAY SYSTEM				
a. Drywell Pressure - High	S	Q	R*(b)	1, 2, 3
b. Containment Pressure - High	S	Q	R*	1, 2, 3
c. Reactor Vessel Water Level - Low, Level 1	S	Q	R*(a)(b)	1, 2, 3
d. Timers				
(1) Subsystem A and B	NA	Q	R	1, 2, 3
(2) Subsystem B	NA	Q	R	1, 2, 3
e. Manual Initiation	NA	R	NA	1, 2, 3
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM				
a. Reactor Vessel Water Level - High, Level 8	S	Q	R*(a)(b)	1
3. SUPPRESSION POOL MAKEUP SYSTEM				
a. Drywell Pressure - High	S	Q	R*(b)	1, 2, 3
b. Reactor Vessel Water Level - Low, Level 1	S	Q	R*(a)(b)	1, 2, 3
c. Suppression Pool Water Level - Low	S	Q	R*	1, 2, 3
d. Suppression Pool Makeup Timer	NA	Q	Q	1, 2, 3
e. SPMU Manual Initiation	NA	R	NA	1, 2, 3

* Calibrate trip unit setpoint at least once per 92 days.
(a) CHANNEL CALIBRATION may be extended to be performed during the fifth refueling outage.
(b) LOGIC SYSTEM FUNCTIONAL TEST may be extended to be performed during the fifth refueling outage.

3/4.4.2 SAFETY VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 Of the following safety/relief valves, the safety valve function of at least 7 valves and the relief valve function of at least 6 valves other than those satisfying the safety valve function requirement shall be OPERABLE with the specified lift settings:

Number of Valves	Function	Setpoint* (psig)
8	Safety	1165 ± 11.6 psi
6	Safety	1180 ± 11.8 psi
5	Safety	1190 ± 11.9 psi
1	Relief	1103 ± 15 psi
9	Relief	1113 ± 15 psi
9	Relief	1123 ± 15 psi

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the safety and/or relief valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
b. With one or more safety/relief valves stuck open, close the stuck open safety/relief valve(s); with suppression pool average water temperature 110°F or greater, place the reactor mode switch in the Shutdown position.
c. With one or more safety/relief valve tail-pipe pressure switches inoperable, restore the inoperable switch(es) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
d. With either relief valve function pressure actuation trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status within 7 days; otherwise, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1.1 The tail-pipe pressure switch for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be 30 ± 5 psig by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 92 days, and a
b. CHANNEL CALIBRATION at least once per 18 months.

* The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

SURVEILLANCE REQUIREMENTS (continued)

4.4.2.1.2 The relief valve function pressure actuation instrumentation shall be demonstrated OPERABLE# by performance of a:

a. CHANNEL FUNCTIONAL TEST, including calibration of the trip unit, at least once per 92 days.
b. CHANNEL CALIBRATION, LOGIC SYSTEM FUNCTIONAL TEST** and simulated automatic operation of the entire system at least once per 18 months.##

** SRV solenoid energization shall be used alternating between the "A" solenoid and the "B" solenoid.

When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated ACTIONs may be delayed for up to 6 hours provided the associated Function maintains Relief initiation capability.

May be extended to be performed during the fifth refueling outage.

SAFETY/RELIEF VALVES LOW-LOW SET FUNCTION

LIMITING CONDITION FOR OPERATION

3.4.2.2 The relief valve function and the low-low set function of the following reactor coolant system safety/relief valves shall be OPERABLE with the following settings:

Valve No.	Low-Low Set Function Setpoint* (psig) ± 15 psi Open	Close	Relief Function Setpoint* (psig) Open	Close
1B21-F051D	1033	926	1103 ± 15 psi	1003 ± 20 psi
1B21-F051C	1073	936	1113 ± 15 psi	1013 ± 20 psi
1B21-F051A	1113	946	1113 ± 15 psi	1013 ± 20 psi
1B21-F051B	1113	946	1113 ± 15 psi	1013 ± 20 psi
1B21-F047F	1113	946	1113 ± 15 psi	1013 ± 20 psi
1B21-F051G	1113	946	1113 ± 15 psi	1013 ± 20 psi

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the relief valve function and/or the low-low set function of one of the above required reactor coolant system safety/relief valves inoperable, restore the inoperable relief valve function and the low-low set function to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the relief valve function and/or the low-low set function of more than one of the above required reactor coolant system safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

c. With either relief valve/low-low set function pressure actuation trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status within 7 days; otherwise, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2.1 The relief valve function and the low-low set function pressure actuation instrumentation shall be demonstrated OPERABLE# by performance of a:

a. CHANNEL FUNCTIONAL TEST, including calibration of the trip unit, at least once per 92 days.

b. CHANNEL CALIBRATION, LOGIC SYSTEM FUNCTIONAL TEST and simulated automatic operation of the entire system at least once per 18 months.##

* The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated ACTIONs may be delayed for up to 6 hours provided the associated Function maintains Low-Low Set initiation capability.

May be extended to be performed during the fifth refueling outage.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.3.1 The following reactor coolant system leakage detection systems shall be OPERABLE:

a. The drywell floor drain sump and equipment drain sump flow monitoring system.

b. Any 2 of the following:

1. Drywell atmosphere particulate radioactivity monitoring system.
2. Drywell atmosphere gaseous radioactivity monitoring system.
3. Upper drywell air coolers condensate flow rate monitoring system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 AND 3.

ACTION:

a. With the drywell floor drain sump or equipment drain sump flow monitoring system inoperable, operation may continue for up to 30 days provided that the upper drywell coolers condensate flow rate monitoring system is one of the two systems OPERABLE per 3.4.3.1.b.

b. With only one of the systems required by 3.4.3.1.b OPERABLE operations may continue for up to 30 days provided:

1. The drywell floor drain sump and equipment drain sump flow monitoring system is OPERABLE.
2. Grab samples of the drywell atmosphere are obtained and analyzed at least once per 24 hours.

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

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

a. Drywell atmosphere particulate and gaseous monitoring systems-performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

b. Drywell floor drain and equipment drain sump flow monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.#

c. Upper drywell air coolers condensate flow rate monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.#

May be extended to be performed during the fifth refueling outage.

SURVEILLANCE REQUIREMENTS

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

a. Monitoring the drywell atmospheric particulate or gaseous radioactivity at least once per 12 hours (not a means of quantifying leakage),

b. Monitoring the drywell floor and equipment sump flow rate at least once per 12 hours.

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

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

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit:

a. At least once per 18 months, ** #

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate, and

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

** P.I.V. leak test extension to the first refueling outage is permissible for each Reactor Coolant System P.I.V. LISTED IN TABLE 3.4.3.2-1, which are identified in letter PY-CEI/NRR-0714L (dated September 11, 1987) as needing a plant outage to test. For this one time test interval, the provisions of Specification 4.0.2 are not applicable.

May be extended to be performed during the fifth refueling outage.

SURVEILLANCE REQUIREMENTS

4.5.1 ECCS division 1, 2 and 3 shall be demonstrated OPERABLE by:

a. At least once per 31 days for the LPCS, LPCI and HPCS systems:

1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

2. Verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct* position.

b. Verifying that, when tested pursuant to Specification 4.0.5, each:

1. LPCS pump develops a flow of at least 6110 gpm at a differential pressure greater than or equal to 128 psid for the system.

2. LPCI pump develops a flow of at least 7100 gpm at a differential pressure greater than or equal to 24 psid for the system.

3. HPCS pump develops a flow of at least 6110 gpm at a differential pressure greater than or equal to 200 psid for the system.

c. For the LPCS, LPCI and HPCS systems, at least once per 18 months:

1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.#

2. Performing a CHANNEL CALIBRATION of the ECCS discharge line "keep filled" pressure alarm instrumentation.

d. For the HPCS system, at least once per 18 months, verifying that the suction is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank low water level signal and on a suppression pool high water level signal.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

May be extended to the completion of the fifth refueling outage.

SURVEILLANCE REQUIREMENTS (Continued)

e. For the ADS by:

1. At least once per 31 days, performing a CHANNEL FUNCTIONAL TEST of the safety related instrument air system low pressure alarm system.

2. At least once per 18 months:

a) Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.#

b) Manually** opening each ADS valve when the reactor steam dome pressure is greater than or equal to 100 psig* and observing that either:

1) The control valve or bypass valve position responds accordingly, or

2) There is a corresponding change in the measured steam flow, or

3) The safety relief valve discharge pressure switch indicates the valve is open.

c) Performing a CHANNEL CALIBRATION of the safety related instrument air system low pressure alarm system and verifying an alarm setpoint of ≥ 155 psig with an allowable value of ≥ 151.9 psig on decreasing pressure.

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* The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

** ADS solenoid energization shall be used alternating between ADS Division 1 and ADS Division 2.

May be extended to be performed during the fifth refueling outage.

SURVEILLANCE REQUIREMENTS (Continued)

1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 L_a. The formula to be used is:

   $[L_o + L_{am} - 0.25\ L_a] \leq L_c \leq [L_o + L_{am} + 0.25L_a]$ where

   L_c = supplemental test result; L_o = superimposed leakage;

   L_{am} = measured Type A leakage.

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

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

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

   1. Air locks,

   2. Main steam line isolation valves,

   3. Valves pressurized with fluid from a seal system,

   4. All containment isolation valves in hydrostatically tested lines which penetrate the primary containment, and

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

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

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

g. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J of 10 CFR 50 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 and the seal system capacity is adequate to maintain system pressure for at least 30 days.

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

* Unless a hydrostatic test is required.

** May be extended to be completed during the fifth refueling outage for valve 1E12F068.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each containment 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 specified isolation time.

4.6.4.2 Each automatic containment isolation valve shall be demonstrated OPERABLE 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 isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

May be extended to the completion of the fifth refueling outage.

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b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem by:

   1. Verifying that the subsystem satisfies the in-place penetration testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, while operating the system at a flow rate of 2000 scfm ± 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a. of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 0.175% when tested at a temperature of 30° and at a relative humidity of 70% in accordance with ASTM D3803; and

   3. Verifying a subsystem flow rate of 2000 scfm ± 10% during system operation when tested in accordance with ANSI N510-1980. The installed air flow monitor can be used to determine flow in lieu of the pitot traverse.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 0.175% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803;

d. At least once per 18 months by:

   1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence for the LOCA.#

   2. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.0 inches water gauge while operating the filter train at a flow rate of 2000 scfm ± 10%.

   3. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

      a. Manual initiation from the control room, and

      b. Simulated automatic initiation signal.#

   4. Verifying that the heaters dissipate 20 kw ± 10% when tested in accordance with ANSI N510-1980.

May be extended to the completion of the fifth refueling outage.

3/4.7 PLANT SYSTEMS

3/4.7.1 COOLING WATER SYSTEMS

EMERGENCY SERVICE WATER SYSTEM (LOOPS A, B, C)

LIMITING CONDITION FOR OPERATION

3.7.1.1 The emergency service water (ESW) loop(s) shall be OPERABLE which is associated with systems or components which are required to be OPERABLE. Each OPERABLE ESW loop shall be comprised of:

a. One OPERABLE ESW pump, and

b. An OPERABLE flow path capable of taking suction from Lake Erie and transferring water through the associated systems and components heat exchanger(s) that are required to be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5 and *.

ACTION:

With an emergency service water loop(s) inoperable which is associated with system(s) or component(s) required to be OPERABLE, declare the associated system(s) or component(s) inoperable and take the ACTION required by the applicable Specification(s).

SURVEILLANCE REQUIREMENTS

4.7.1.1 The above required emergency service water system loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 18 months# during shutdown by verifying that each automatic valve servicing safety related equipment actuates to the correct position on a LOCA test signal.

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

May be extended to the completion of the fifth refueling outage.

3/4.7 PLANT SYSTEMS

EMERGENCY CLOSED COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 The emergency closed cooling (ECC) loop(s) shall be OPERABLE which is associated with systems or components which are required to be OPERABLE. Each OPERABLE ECC loop shall be comprised of:

a. One OPERABLE ECC pump, and

b. An OPERABLE flow path capable of transferring water through the associated systems and components heat exchanger(s) that are required to be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5.

ACTION:

With an emergency closed cooling loop(s) inoperable which is associated with system(s) or component(s) required to be OPERABLE, declare the associated system(s) or component(s) inoperable and take the ACTION required by the applicable Specification(s).

SURVEILLANCE REQUIREMENTS

4.7.1.2 The above required emergency closed cooling loop(s) shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 18 months# during shutdown by verifying that each automatic valve servicing safety related equipment actuates to the correct position on a LOCA test signal.

May be extended to the completion of the fifth refueling outage.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem by:

   1. Verifying that the subsystem satisfies the in-place penetration testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 30000 scfm ± 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978 by showing a methyl iodide penetration of less than 1% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803; and

   3. Verifying a subsystem flow rate of 30000 scfm ± 10% during subsystem operation when tested in accordance with ANSI N510-1980. The installed air flow monitor can be used to determine flow in lieu of a pitot traverse.

d. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 1% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803.

e. At least once per 18 months by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 4.9 inches water gauge while operating the subsystem at a flow rate of 30000 scfm ± 10%.

   2. Verifying that on each of the below emergency recirculation mode actuation test signals, the subsystem automatically switches to the emergency recirculation mode of operation and the isolation dampers close within 10 seconds:

      a) High Drywell Pressure#
      b) Low Reactor Water Level-Level 1#
      c) High radiation from control room ventilation duct

May be extended to the completion of the fifth refueling outage.

3/4.7.4 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.4 All snubbers shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3. OPERATIONAL CONDITIONS 4 and 5 for snubbers located on systems required OPERABLE in those OPERATIONAL CONDITIONS.

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.4.g on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.4 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

A visual inspection of all snubbers shall be performed according to the schedule determined by Table 4.7.4.1. Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently. The visual inspection for each type of snubber shall be determined based on the criteria provided in Table 4.7.4-1 and the initial inspection interval utilizing this criteria shall be 18-months, beginning from the conclusion of the last visual inspection conducted during RF04.#

May be extended to the completion of the fifth refueling outage.

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown#, a representative sample of snubbers shall be tested using one of the following sample plans for each type of snubber. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The Nuclear Regulatory Commission shall be notified in writing pursuant to 10 CFR 50.4 of the sample plan selected prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.4.f., an additional 5% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7.4-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.4.f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7.4-1. If at any time the point plotted falls on or above the "Reject" line all snubbers of that type shall be functionally tested. If at any time the point plotted falls on or below the "Accept" line, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of equipment failure are retested; or

3) An initial representative sample of 55 snubbers of each type shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, $1 + C/2$, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation $N = 55(1 + C/2)$. Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls on or below the "Accept" line or all the snubbers of that type have been tested.

May be extended to the completion of the fifth refueling outage.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments and indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months# during shutdown by transferring unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each of the above required diesel generators shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8.1.1.2-1 on a STAGGERED TEST BASIS by:

   1. Verifying the fuel level in the day tank.

   2. Verifying the fuel level in the fuel storage tank.

   3. Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day tank.

   4. Verifying the diesel starts from ambient conditions and accelerates to at least 441 rpm for Div 1 and Div 2 and 882 rpm for Div 3 in less than or equal to 10 seconds*. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds* after the start signal for Div 1 and Div 2 and 13 seconds* after the start signal for Div 3.

   5. Verifying the diesel generator is synchronized, loaded to between 5600 and 5800 kw** for diesel generators Div 1 and Div 2 and loaded to greater than or equal to 2600 kw for diesel generator Div 3 in less than or equal to 60 seconds*, and operates with this load for at least 60 minutes.

   6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

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* All diesel generator starts for the purpose of this Surveillance Requirement may be preceded by an engine prelube period. The diesel generator start (10 sec)/load (60 sec) from ambient conditions shall be performed at least once per 184 days in these surveillance tests. All other engine starts for the purpose of this surveillance testing may be preceded by other warmup procedures recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test; the loads, however, shall not be less than 5600 kw nor greater than 7000 kw.

May be extended to the completion of the fifth refueling outage.

7. Verifying the pressure in all air start receivers for each diesel generator to be greater than or equal to 210 psig.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day tank.

c. At least once per 92 days by checking for and removing accumulated water from the fuel oil storage tanks.

d. By sampling new fuel oil in accordance with ASTM D4057-88 prior to the addition to the storage tank and:

1. By verifying prior to addition to the storage tanks that the sample has:

a) An API Gravity of within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate; or an absolute specific gravity at 60/60°F, of greater than or equal to 0.83 but less than or equal to 0.89; or an API gravity at 60°F of greater than or equal to 26 degrees but less than or equal to 39 degrees, when tested in accordance with ASTM D1298-85,

b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, when testing in accordance with the tests specified in ASTM D975-89, if gravity was not determined by comparison with the supplier's certification.

c) A flash point equal to or greater than 125°F, when tested in accordance with the tests specified in ASTM D975-89,

d) No visible free water or particulate contamination when tested in accordance with ASTM D4176-86.

2. By verifying within 31 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975-89 are met when tested in accordance with the tests specified in ASTM D975-89.

e. At least once every 31 days by obtaining a sample of fuel oil from the storage tanks in accordance with ASTM D2276-88, and verifying that total particulate contamination is less than 10 mg/liter when tested in accordance with ASTM D2276-88.

f. At least once per 18 months*, **, # during shutdown, by:

1. Subjecting the diesel to an inspection in accordance with instructions prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2. Verifying the diesel generator capability to reject a load of greater than or equal to 1400 kw (LPCS pump) for diesel generator Div 1, greater than or equal to 729 kw (RHR B pump or RHR C pump)

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* For any start of a diesel, the diesel must be loaded in accordance with the manufacturer's recommendations.

** Except 4.8.1.1.2.f.1 to be performed every refueling outage, for the Div 1 and Div 2 diesel generators.

May be extended to the completion of the fifth refueling outage.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 75 TO FACILITY OPERATING LICENSE NO. NPF-58

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-440

1.0 INTRODUCTION

By letters dated March 24, June 9, and June 30, 1995, The Cleveland Electric Illuminating Company, et al. (licensees), proposed an amendment to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant, Unit No. 1 (PNPP). The amendment would permit a one-time extension of the performance intervals for certain current Technical Specification Surveillance Requirements (SRs). Affected SRs include penetration leak rate testing, valve operability testing, instrument calibration, response time testing, and logic system functional tests. The Improved Technical Specifications (ITS), issued on June 23, 1995, are unaffected by this request because the SR extensions will expire prior to the scheduled ITS implementation.

The lengths of the last refueling outage and the present fuel cycle will cause several SRs to become due prior to the next refueling outage (RFO-5) scheduled to begin no later than February 15, 1996. Performance of the SRs under conditions other than a plant shutdown either is not possible or would cause the plant to be placed in an undesirable configuration which may increase the probability of a plant trip.

Some SRs whose OPERABILITY requirements include Operational Conditions 4 and 5 are scheduled to be performed during the system outage windows that were established to provide "defense in depth." Performance of those SRs prior to scheduled system outage windows would result in extension of critical system outage times, and a corresponding potential increase in accident risk during shutdown.

2.0 EVALUATION

In a NRC Safety Evaluation (SE) dated August 2, 1993, the extension of the Peach Bottom Atomic Power Station, Unit Nos. 2 and 3 surveillance intervals from 18 to 24 months were evaluated. The one-time surveillance interval extensions that Perry is requesting are less than the six months granted in the Peach Bottom SE. The longest surveillance interval extension that Perry is requesting to the latest scheduled beginning of the refueling outage is 51 days. The longest extension that Perry is requesting for certain surveillances to the scheduled end of the refueling outage is 159 days.

Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group, show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability.

With regard to surveillance extensions for Rosemount transmitters, the NRC in its Peach Bottom SE accepted the report, "30 Month Stability Specification For Rosemount Model 1152, 1153, 1154 Pressure Transmitters." That report supported the extension of the calibration interval for the transmitters from 18 to 30 months based on a reduction in the drift allowance. The existing PNPP setpoint calculations for Rosemount transmitters and associated trip unit channels are bounding, and PNPP has adequate allowance in its calculations for 30 month transmitter drift.

The PNPP reactor protection system (RPS) has redundancy, diversity, and independent trip systems such that a single failure will neither cause nor prevent a required reactor scram. Also, instrumentation failure is a small fraction of the scram failure probability. Therefore, a one-time extension of the RPS response time surveillance intervals is acceptable.

The emergency core cooling system (ECCS) instrumentation response times were reviewed by the licensee, and it determined that there were no failure modes which will affect the response time of the instrumentation loop which would not be detected by other surveillances such as channel calibration, channel functional tests, or other techniques. Based on the above, and the redundancy and diversity of the ECCS instrumentation, the NRC staff finds that a one-time extension of the ECCS response time surveillance intervals is acceptable.

ECCS undervoltage, degraded voltage, and timing relays are also proposed to have their surveillances extended. An analysis of drift data for the relays shows that the relay setpoints should remain within their allowable value range for the proposed extension period. Therefore, a one-time extension of the applicable relay SRs is acceptable.

Also, proposed for extension are some offsite and onsite electrical circuit SRs. These SRs involve normal to alternate circuit transfer and diesel generator testing. Four offsite power sources are available to provide power to the 4.16 kV emergency busses. The failure of any one offsite power source does not result in a total loss of offsite power to the bus. The diesel generators are tested monthly and quarterly to show operability. There have been no failures of the diesel generators to start and run since March 1991, and the last 2 performances of the 18 month SRs for the diesel generators resulted in no adverse findings. Therefore, a one-time extension of the 18 month SRs for the diesel generators and transfer circuit testing is acceptable.

Some radiation and leakage detection monitors are also included in the request for surveillance extension. The NRC staff has reviewed the provided monitors drift data information and finds it acceptable for granting a one-time extension of the monitors surveillance intervals.

Some containment and reactor coolant pressure isolation valves and a reactor core isolation cooling turbine exhaust valve were also proposed to have their leakage testing extended. The valves exhibited either no leakage or leakage significantly lower than the limits during their last test; therefore, a one-time extension for the next leakage test is acceptable. Some containment isolation, emergency closed cooling water, and emergency service water valves were proposed to have their automatic actuation SR extended. These valves have redundancy in system performance and are periodically tested in a different manner during power operation. Therefore, a one-time SR extension is acceptable. Additionally, some radiation monitor, containment, suppression pool, and air valves are proposed to have their position indication tests extended. The last two surveillances for these position indications had no failures. Therefore, a one-time extension of the SRs is acceptable.

Some snubber testing is also proposed for extension. Snubber testing is an overall part of a ten year inservice testing program. The 18 month testing technical specification requirement is mainly to ensure that testing is conducted throughout the ten year period instead of during one discrete time interval. Therefore, a one-time extension of snubber testing to accommodate the projected refueling outage start and end dates is acceptable.

The annulus exhaust gas treatment system and the control room emergency recirculation system are also proposed to have their 18 month SRs extended into RFO5. These systems are tested for operation every 31 days. Because the systems mechanical operation is assured, a one-time extension of the 18 month SR, that tests the complete systems logic, is acceptable.

The NRC staff has reviewed the licensee's request for SR extensions in order to reach and perform the fifth refueling outage in a planned manner including safe shutdown considerations, and based on the above, finds the proposed amendment acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or a change to a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding (60 FR 24919 and 60 FR 42612). Accordingly, this 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 this amendment.

5.0 CONCLUSION

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

Principal Contributor: J. Hopkins

Date: November 29, 1995