Docket File

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

December 2, 1991

Docket Nos. 50-352
and 50-353

Mr. George J. Beck
Manager-Licensing, MC 5-2A-5
Philadelphia Electric Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, Pennsylvania 19087-0195

Dear Mr. Beck:

SUBJECT: REDUCED TESTING OF RPS, ECCS AND COMMON INSTRUMENTATION, CHANGE REQUEST 89-16, LIMERICK GENERATING STATION, UNITS 1 AND 2 (TAC NOS. M76689 AND M76690)

The Commission has issued the enclosed Amendment No. 53 to Facility Operating License No. NPF-39 and Amendment No. 17 to Facility Operating License No. NPF-85 for the Limerick Generating Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in partial response to your application dated April 26, 1990.

These amendments revise the TSs to extend the surveillance test intervals (STIs) and allowable out-of-service times (AOTs) for instrumentation supporting the Reactor Protection System (RPS) and Emergency Core Cooling System (ECCS), including instrumentation common to the Control Rod Block Function (CRBF), the Reactor Core Isolation Cooling (RCIC) system, and the isolation instrumentation common to RPS and/or ECCS.

As I discussed with your staff, there has been a delay in processing this application, since some of the proposed changes were not covered by previous NRC safety evaluations. In your role as former chairman of the BWR Owners Group, you submitted by letter dated February 19, 1991 two topical reports, which provided additional analyses to support changes to the STIs and AOTs for the RCIC instrumentation and the End-of-Cycle Recirculation Pump Trip (EOC-RPT) instrumentation. Our letter of September 13, 1991 informed you that we found that GENE-770-06-2 provided an acceptable basis for extending STIs and AOTs for the RCIC actuation instrumentation. We have not completed our review of the topical report on the EOC-RPT system. However, rather than delay this amendment application any longer, we are issuing these amendments on everything in your application except the few TS pages related to the EOC-RPT. The latter will be handled as a separate action.

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

Sincerely,

Original signed by
Richard J. Clark

Richard J. Clark, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 53 to License No. NPF-39
   Amendment No. 17 to License No. NPF-85
2. Safety Evaluation

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

Sincerely,

Richard J. Clark

Richard J. Clark, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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cc w/enclosures:
See next page

Mr. George J. Beck
Philadelphia Electric Company

Limerick Generating Station
Units 1 & 2

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

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-352

LIMERICK GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 53
License No. NPF-39

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

   A. The application for amendment by Philadelphia Electric Company (the licensee) dated April 26, 1990, 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-39 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 53 , are hereby incorporated into this license. Philadelphia Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective fifteen (15) days after date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard J. Clark for

Charles L. Miller, Director
Project Directorate I-2
Division of Reactor Projects - I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 2, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 53

FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

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. Overleaf pages are provided to maintain document completeness.*

Remove	Insert
xix	xix
xx	xx*
3/4 3-1	3/4 3-1
3/4 3-2	3/4 3-2*
3/4 3-5	3/4 3-5
3/4 3-6	3/4 3-6*
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
3/4 3-9	3/4 3-9
3/4 3-10	3/4 3-10
3/4 3-15	3/4 3-15*
3/4 3-16	3/4 3-16
3/4 3-17	3/4 3-17
3/4 3-18	3/4 3-18*
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29*
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 3-32	3/4 3-32*
3/4 3-35	3/4 3-35
3/4 3-36	3/4 3-36
3/4 3-39	3/4 3-39*
3/4 3-40	3/4 3-40
3/4 3-41	3/4 3-41
3/4 3-42	3/4 3-42*

ATTACHMENT TO LICENSE AMENDMENT NO. 53

FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

Remove	Insert
3/4 3-53	3/4 3-53
3/4 3-54	3/4 3-54
3/4 3-55	3/4 3-55*
3/4 3-56	3/4 3-56
3/4 3-61	3/4 3-61
3/4 3-62	3/4 3-62*
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 3-4	B 3/4 3-4
B 3/4 3-5	B 3/4 3-5
B 3/4 3-6	B 3/4 3-6*

BASES

SECTION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within 12 hours. The provisions of Specification 3.0.4 are not applicable.

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

SURVEILLANCE REQUIREMENTS

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

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

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

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* An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 6 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

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

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors[b]:			
a. Neutron Flux - High	2	3	1
	3, 4	3	2
	5(c)	3(d)	3
b. Inoperative	2	3	1
	3, 4	3	2
	5	3(d)	3
2. Average Power Range Monitor[e]:			
a. Neutron Flux - Upscale, Setdown	2	2	1
	3	2	2
	5(c)(l)	2(d)	3
b. Neutron Flux - Upscale			
1) Flow Biased	1	2	4
2) High Flow Clamped	1	2	4
c. Inoperative	1, 2	2	1
	3	2	2
	5(c)(l)	2(d)	3
d. Downscale	1(g)	2	4
3. Reactor Vessel Steam Dome Pressure - High	1, 2(f)	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Main Steam Line Isolation Valve - Closure	1(g)	1/valve	4

TABLE NOTATIONS

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

(b) This function shall be automatically bypassed when the reactor mode switch is in the Run position and the associated APRM is not downscale.

(c) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations performed per Specification 3.10.3.

(d) The noncoincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per Trip System is 4 APRMs, 6 IRMs and 2 SRMs.

(e) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

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

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

(h) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

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

(j) This function shall be automatically bypassed when turbine first stage pressure is equivalent to a THERMAL POWER of less than 30% of RATED THERMAL POWER.

(k) Also actuates the EOC-RPT system.

(l) Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	N.A.
b. Inoperative	N.A.
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale, Setdown	N.A.
b. Neutron Flux - Upscale	
1) Flow Biased	≤0.09
2) High Flow Clamped	≤0.09
c. Inoperative	N.A.
d. Downscale	N.A.
3. Reactor Vessel Steam Dome Pressure - High	≤ 0.55
4. Reactor Vessel Water Level - Low, Level 3	≤ 1.05
5. Main Steam Line Isolation Valve - Closure	≤ 0.06
6. Main Steam Line Radiation - High	N.A.
7. Drywell Pressure - High	N.A.
8. Scram Discharge Volume Water Level - High	
a. Level Transmitter	N.A.
b. Float Switch	N.A.
9. Turbine Stop Valve - Closure	≤ 0.06
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≤ 0.08**
11. Reactor Mode Switch Shutdown Position	N.A.
12. Manual Scram	N.A.

*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.

**Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U,S(b)	S/U(c), W	R	2
	S	W(j)	R	3, 4, 5
b. Inoperative	N.A.	W(j)	N.A.	2, 3, 4, 5
2. Average Power Range Monitor(f):				
a. Neutron Flux - Upscale, Setdown	S/U,S(b)	S/U(c), W	SA	2
	S	W(j)	SA	3, 5(k)
b. Neutron Flux - Upscale				
1) Flow Biased	S,D(g)	S/U(c), Q	W(d)(e), SA	1
2) High Flow Clamped	S	S/U(c), Q	W(d)(e), SA	1
c. Inoperative	N.A.	Q(j)	N.A.	1, 2, 3, 5(k)
d. Downscale	S	Q	SA	1
3. Reactor Vessel Steam Dome Pressure - High	S	Q	R	1, 2(h)
4. Reactor Vessel Water Level - Low, Level 3	S	Q	R	1, 2
5. Main Steam Line Isolation Valve - Closure	N.A.	Q	R	1
6. Main Steam Line Radiation - High	S	Q	R	1, 2(h)
7. Drywell Pressure - High	S	Q	R	1, 2
8. Scram Discharge Volume Water Level - High				
a. Level Transmitter	S	Q	R	1, 2, 5(i)
b. Float Switch	N.A.	Q	R	1, 2, 5(i)

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
9. Turbine Stop Valve - Closure	N.A.	Q	R	1
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	N.A.	Q	R	1
11. Reactor Mode Switch Shutdown Position	N.A.	R	N.A.	1, 2, 3, 4, 5
12. Manual Scram	N.A.	W	N.A.	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for a least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.
(c) Within 24 hours prior to startup, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.
(g) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing loop flow (APRM % flow). During the startup test program, data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon the conclusion of the startup test program.
(h) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.
(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
(j) If the RPS shorting links are required to be removed per Specification 3.9.2, they may be reinstalled for up to 2 hours for required surveillance. During this time, CORE ALTERATIONS shall be suspended, and no control rod shall be moved from its existing position.
(k) Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3.

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 channels(s) in the tripped condition would cause an isolation, the inoperable channel(s) shall be restored to OPERABLE status within:

a) 2 hours for trip functions not common* to the Reactor Protection System (RPS) and/or Emergency Core Cooling System (ECCS) Actuation Instrumentation, or

b) 6 hours for trip functions common* to RPS and/or ECCS Actuation Instrumentation.

If this cannot be accomplished, the ACTION required by Table 3.3.2-1 for the affected trip function shall be taken, or the channel shall be placed in the tripped condition.

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) 1 hour for trip functions not common* to the RPS and/or ECCS Actuation Instrumentation,

b) 12 hours for trip functions common* to RPS Instrumentation,

c) 24 hours for trip functions common* to ECCS Actuation Instrumentation,

and

d) 12 hours for trip functions common* to RPS and ECCS Actuation Instrumentation.

The provisions of Specification 3.0.4 are not applicable.

* Trip functions common to RPS and/or ECCS Actuation Instrumentation are shown in Table 4.3.2.1-1.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

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 1 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 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.

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

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL (a),(c)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
7. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level Low, Low - Level 2	B	2	1, 2, 3	25
b. Drywell Pressure - High	H	2	1, 2, 3	25
c.1. Refueling Area Unit 1 Ventilation Exhaust Duct Radiation - High	R	2	*#	25
2. Refueling Area Unit 2 Ventilation Exhaust Duct Radiation - High	R	2	*#	25
d. Reactor Enclosure Ventilation Exhaust Duct Radiation - High	S	2	1, 2, 3	25
e. Outside Atmosphere To Reactor Enclosure Δ Pressure - Low	U	1	1, 2, 3	25
f. Outside Atmosphere To Refueling Area Δ Pressure - Low	T	1	*	25
g. Reactor Enclosure Manual Initiation	NA	1	1, 2, 3	24
h. Refueling Area Manual Initiation	NA	1	*	25

Amendment No. 6, 23, 33, 40
MAY 3 0 1990

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

ACTION 21 - 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 22 - Be in at least STARTUP within 6 hours.

ACTION 23 - In OPERATIONAL CONDITION 1 or 2, verify the affected system isolation valves are closed within 1 hour and declare the affected system inoperable. In OPERATIONAL CONDITION 3, be in at least COLD SHUTDOWN within 12 hours.

ACTION 24 - 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 or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 25 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

ACTION 26 - Close the affected system isolation valves within 1 hour.

TABLE NOTATIONS

* Required when (1) handling irradiated fuel in the refueling area secondary containment, or (2) during CORE ALTERATIONS, or (3) during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

** May be bypassed under administrative control, with all turbine stop valves closed.

\# During operation of the associated Unit 1 or Unit 2 ventilation exhaust system.

(a) See Specification 3.6.3, Table 3.6.3-1 for primary containment isolation valves which are actuated by these isolation signals.

(b) A channel may be placed in an inoperable status for up to:

a) 2 hours for trip functions not common the the Reactor Protection System (RPS) and/or Emergency Core Cooling System (ECCS) Actuation Instrumentation, or
b) 6 hours for trip functions common to RPS and/or ECCS Actuation Instrumentation

for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter. Trip functions common to RPS and/or ECCS Actuation Instrumentation are shown in Table 4.3.2.1-1. In addition, for the HPCI system and RCIC system isolation, provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is OPERABLE and all required actuation instrumentation for that valve is OPERABLE, one channel may be placed in an inoperable status for up to 8 hours for required surveillance without placing the channel or trip system in the tripped condition.

TABLE NOTATIONS

(c) Actuates secondary containment isolation valves shown in Table 3.6.5.2.1-1 and/or 3.6.5.2.2-1 and signals B, H, S, U, R and T also start the standby gas treatment system.

(d) RWCU system inlet outboard isolation valve closes on SLCS "B" initiation. RWCU system inlet inboard isolation valve closes on SLCS "A" or SLCS "C" initiation.

(e) Manual initiation isolates the steam supply line outboard isolation valve and only following manual or automatic initiation of the system.

(f) In the event of a loss of ventilation the temperature - high setpoint may be raised by 50°F for a period not to exceed 30 minutes to permit restoration of the ventilation flow without a spurious trip. During the 30 minute period, an operator, or other qualified member of the technical staff, shall observe the temperature indications continuously, so that, in the event of rapid increases in temperature, the main steam lines shall be manually isolated.

(g) Wide range accident monitor per Specification 3.3.7.5.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Low - Level 2	≥ - 38 inches*	≥ - 45 inches
2) Low, Low, Low - Level 1	≥ - 129 inches*	≥ - 136 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	≥ 756 psig	≥ 736 psig
d. Main Steam Line Flow - High	≤ 108.7 psid	≤ 111.7 psid
e. Condenser Vacuum - Low	10.5 psia	≥ 10.1 psia/≤ 10.9 psia
f. Outboard MSIV Room Temperature - High	≤ 192°F	≤ 200°F
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	≤ 165°F	≤ 175°F
h. Manual Initiation	N.A.	N.A.
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
a. Reactor Vessel Water Level Low - Level 3	≥ 12.5 inches*	≥ 11.0 inches
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 75 psig	≤ 95 psig
c. Manual Initiation	N.A.	N.A.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHIC SURVEILLANCE REQUIR
1. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level##				
1) Low, Low, Level 2	S	Q	R	1, 2, 3
2) Low, Low, Low - Level 1	S	Q	R	1, 2, 3
b. Main Steam Line Radiation### - High	S	Q	R	1, 2, 3
c. Main Steam Line Pressure - Low	S	M	R	1
d. Main Steam Line Flow - High	S	M	R	1, 2, 3
e. Condenser Vacuum - Low	S	M	R	1, 2**, 3**
f. Outboard MSIV Room Temperature - High	S	M	R	1, 2, 3
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	S	M	R	1, 2, 3
h. Manual Initiation	N.A.	R	N.A.	1, 2, 3
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level### Low - Level 3	S	Q	R	1, 2, 3
b. Reactor Vessel (RHR Cut-In Permissive) Pressure## - High	S	Q	R	1, 2, 3
c. Manual Initiation	N.A.	R	N.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 FOR WHICH SURVEILLANCE REQUIRED
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. RWCS Δ Flow - High	S	M	R	1, 2, 3
b. RWCS Area Temperature - High	S	M	R	1, 2, 3
c. RWCS Area Ventilation Δ Temperature - High	S	M	R	1, 2, 3
d. SLCS Initiation	N.A.	R	N.A.	1, 2, 3
e. Reactor Vessel Water Level## Low, Low, - Level 2	S	Q	R	1, 2, 3
f. Manual Initiation	N.A.	R	N.A.	1, 2, 3
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Δ Pressure - High	S	M	R	1, 2, 3
b. HPCI Steam Supply Pressure - Low	S	M	R	1, 2, 3
c. HPCI Turbine Exhaust Diaphragm Pressure - High	S	M	R	1, 2, 3
d. HPCI Equipment Room Temperature - High	S	M	R	1, 2, 3
e. HPCI Equipment Room Δ Temperature - High	S	M	R	1, 2, 3
f. HPCI Pipe Routing Area Temperature - High	S	M	R	1, 2, 3
g. Manual Initiation	N.A.	R	N.A.	1, 2, 3
h. HPCI Steam Line Δ Pressure Timer	N.A.	M	R	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 FOR WHICH SURVEILLANCE REQUIRED
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Δ Pressure - High	S	M	R	1, 2, 3
b. RCIC Steam Supply Pressure - Low	S	M	R	1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure - High	S	M	R	1, 2, 3
d. RCIC Equipment Room Temperature - High	S	M	R	1, 2, 3
e. RCIC Equipment Room Δ Temperature - High	S	M	R	1, 2, 3
f. RCIC Pipe Routing Area Temperature - High	S	M	R	1, 2, 3
g. Manual Initiation	N.A.	R	N.A.	1, 2, 3
h. RCIC Steam Line Δ Pressure Timer	N.A.	M	R	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 FOR WHICH SURVEILLANCE REQUIR
6. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level##				
1) Low, Low - Level 2	S	Q	R	1, 2, 3
2) Low, Low, Low - Level 1	S	Q	R	1, 2, 3
b. Drywell Pressure### - High	S	Q	R	1, 2, 3
c. North Stack Effluent Radiation - High	S	Q	R	1, 2, 3
d. Deleted				
e. Reactor Enclosure Ventilation Exhaust Duct - Radiation - High	S	M	R	1, 2, 3
f. Outside Atmosphere to Reactor Enclosure Δ Pressure - Low	N.A.	M	Q	1, 2, 3
g. Deleted				
h. Drywell Pressure## - High/ Reactor Pressure## - Low	S	Q	R	1, 2, 3
i. Primary Containment Instrument Gas to Drywell Δ Pressure - Low	N.A.	M	Q	1, 2, 3
j. Manual Initiation	N.A.	R	N.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 FOR WHICH SURVEILLANCE REQUIRED
7. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level## Low, Low - Level 2	S	Q	R	1, 2, 3
b. Drywell Pressure### - High	S	Q	R	1, 2, 3
c. 1. Refueling Area Unit 1 Ventilation Exhaust Duct Radiation - High	S	M	R	*#
2. Refueling Area Unit 2 Ventilation Exhaust Duct Radiation - High	S	M	R	*#
d. Reactor Enclosure Ventilation Exhaust Duct Radiation - High	S	M	R	1, 2, 3
e. Outside Atmosphere To Reactor Enclosure Δ Pressure - Low	N.A.	M	Q	1, 2, 3
f. Outside Atmosphere To Refueling Area Δ Pressure - Low	N.A.	M	Q	*
g. Reactor Enclosure Manual Initiation	N.A.	R	N.A.	1, 2, 3
h. Refueling Area Manual Initiation	N.A.	R	N.A.	*

*Required when (1) handling irradiated fuel in the refueling area secondary containment, or (2) during CORE ALTERATIONS, or (3) during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

**When not administratively bypassed and/or when any turbine stop valve is open.

#During operation of the associated Unit 1 or Unit 2 ventilation exhaust system.

##These trip functions (1a, 2b, 3e, 6a, 6h, and 7a) are common to the ECCS actuation trip function.

###These trip functions (2a, 6b, and 7b) are common to the RPS and ECCS actuation trip functions.

####This trip function (1b) is common to the RPS trip function.

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 subsystem inoperable, restore the inoperable trip system to OPERABLE status within:

   1. 7 days, provided that the HPCI and RCIC systems are OPERABLE.

   2. 72 hours.

   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.

TABLE NOTATIONS

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

(b) Also provides input to actuation logic for the associated emergency diesel generators.

(c) One trip system. Provides signal to HPCI pump suction valves only.

(d) On 1 out of 2 taken twice logic, provides a signal to trip the HPCI pump turbine only.

(e) The manual initiation push buttons start the respective core spray pump and diesel generator. The "A" and "B" logic manual push buttons also actuate an initiation permissive in the injection valve opening logic.

(f) A channel as used here is defined as the 127 bus relay for Item 1 and the 127, 127Y, and 127Z feeder relays with their associated time delay relays taken together for Item 2.

* When the system is required to be OPERABLE per Specification 3.5.2.

Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

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

Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.

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 inoperable.

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

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

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within 24 hours.

ACTION 33 - 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 ECCS inoperable.

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. For one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or declare the HPCI system inoperable.

b. With more than one channel inoperable, declare the HPCI system inoperable.

ACTION 35 - 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 HPCI system inoperable.

ACTION 36 - With the number of OPERABLE channels less than the Total Number of Channels, declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2, as appropriate.

ACTION 37 - With the number of OPERABLE channels one less than the Total Number of Channels, place the inoperable channel in the tripped condition within 1 hour; operation may then continue until performance of the next required CHANNEL FUNCTIONAL TEST.

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

ECCS	RESPONSE TIME (Seconds)
1. CORE SPRAY SYSTEM	$\leq$ 27
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM	$\leq$ 40
3. AUTOMATIC DEPRESSURIZATION SYSTEM	N.A.
4. HIGH PRESSURE COOLANT INJECTION SYSTEM	$\leq$ 30
5. LOSS OF POWER	N.A.

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
1. CORE SPRAY SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	Q	R	1, 2, 3
c. Reactor Vessel Pressure - Low	S	Q	R	1, 2, 3, 4*, 5*
d. Manual Initiation	N.A.	R	N.A.	1, 2, 3, 4*, 5*
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	Q	R	1, 2, 3
c. Reactor Vessel Pressure - Low	S	Q	R	1, 2, 3
d. Injection Valve Differential Pressure - Low (Permissive)	S	Q	R	1, 2, 3, 4*, 5*
e. Manual Initiation	N.A.	R	N.A.	1, 2, 3, 4*, 5*
3. HIGH PRESSURE COOLANT INJECTION SYSTEM***				
a. Reactor Vessel Water Level - Low Low, Level 2	S	Q	R	1, 2, 3
b. Drywell Pressure - High	S	Q	R	1, 2, 3
c. Condensate Storage Tank Level - Low	S	Q	R	1, 2, 3
d. Suppression Pool Water Level - High	S	Q	R	1, 2, 3
e. Reactor Vessel Water Level - High, Level 8	S	Q	R	1, 2, 3
f. Manual Initiation	N.A.	R	N.A.	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
4. AUTOMATIC DEPRESSURIZATION SYSTEM#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	R	1, 2, 3
b. Drywell Pressure - High	S	Q	R	1, 2, 3
c. ADS Timer	N.A.	Q	Q	1, 2, 3
d. Core Spray Pump Discharge Pressure - High	S	Q	R	1, 2, 3
e. RHR LPCI Mode Pump Discharge Pressure - High	S	Q	R	1, 2, 3
f. Reactor Vessel Water Level - Low, Level 3	S	Q	R	1, 2, 3
g. Manual Initiation	N.A.	R	N.A.	1, 2, 3
h. ADS Drywell Pressure Bypass Timer	N.A.	Q	Q	1, 2, 3
5. LOSS OF POWER				
a. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)##	N.A.	R	N.A.	1, 2, 3, 4**, 5**
b. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	S	M	R	1, 2, 3, 4**, 5**

* When the system is required to be OPERABLE per Specification 3.5.2.

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

*** Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

\## Loss of Voltage Relay 127-11X is not field setable.

INSTRUMENTATION

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With an ATWS recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.1-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within 1 hour.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:

    1. If the inoperable channels consist of one reactor vessel water level channel and one reactor vessel pressure channel, place both inoperable channels in the tripped condition within 1 hour, or, if this action will initiate a pump trip, declare the trip system inoperable.

    2. If the inoperable channels include two reactor vessel water level channels or two reactor vessel pressure channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within 1 hour or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

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

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

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION*	ACTION
a. Reactor Vessel Water Level - Low Low, Level 2	4#	50
b. Reactor Vessel Water Level - High, Level 8	4#	51
c. Condensate Storage Tank Water Level - Low	2**	52
d. Manual Initiation	1/system***	53

*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 all other channels monitoring that parameter are OPERABLE.

**One trip system with one-out-of-two logic.

***One trip system with one channel.

#One trip system with one-out-of-two twice logic.

REACTOR CORE ISOLATION COOLING SYSTEM
ACTION STATEMENTS

ACTION 50 - 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 RCIC system inoperable.

b. With more than one channel inoperable, declare the RCIC system inoperable.

ACTION 51 - With the number of OPERABLE channels less than required by the minimum OPERABLE channels per Trip System requirement, declare the RCIC system inoperable within 24 hours.

ACTION 52 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, place at least one inoperable channel in the tripped condition within 24 hours or declare the RCIC system inoperable.

ACTION 53 - With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the inoperable channel to OPERABLE status within 24 hours or declare the RCIC system inoperable.

TABLE 3.3.5-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNITS		TRIP SETPOINT	ALLOWABLE VALUE
a.	Reactor Vessel Water Level - Low Low, Level 2	$\geq$-38 inches*	$\geq$-45 inches
b.	Reactor Vessel Water Level - High, Level 8	$\leq$ 54 inches	$\leq$ 60 inches
c.	Condensate Storage Tank Level - Low	$\geq$ 135.8** inches	$\geq$ 132.3 inches
d.	Manual Initiation	N.A.	N.A.

*See Bases Figure B 3/4.3-1.

**Corresponds to 2.3 feet indicated.

TABLE 4.3.5.1-1

REACTOR CORE ISOLATION SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level - Low Low, Level 2	S	Q	R
b. Reactor Vessel Water Level - High, Level 8	S	Q	R
c. Condensate Storage Tank Level - Low	S	Q	R
d. Manual Initiation	N.A.	R	N.A.

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 BLOCK MONITOR				
a. Upscale	N.A.	S/U(b)(c),Q(c)	SA	1*
b. Inoperative	N.A.	S/U(b)(c),Q(c)	N.A.	1*
c. Downscale	N.A.	S/U(b)(c),Q(c)	SA	1*
2. APRM				
a. Flow Biased Neutron Flux - Upscale	N.A.	S/U(b),Q	SA	1
b. Inoperative	N.A.	S/U(b),Q	N.A.	1, 2, 5***
c. Downscale	N.A.	S/U(b),Q	SA	1
d. Neutron Flux - Upscale, Startup	N.A.	S/U(b),Q	SA	2, 5***
3. SOURCE RANGE MONITORS				
a. Detector not full in	N.A.	S/U(b),W	N.A.	2, 5
b. Upscale	N.A.	S/U(b),W	SA	2, 5
c. Inoperative	N.A.	S/U(b),W	N.A.	2, 5
d. Downscale	N.A.	S/U(b),W	SA	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	N.A.	S/U(b),W	N.A.	2, 5
b. Upscale	N.A.	S/U(b),W	SA	2, 5
c. Inoperative	N.A.	S/U(b),W	N.A.	2, 5
d. Downscale	N.A.	S/U(b),W	SA	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	N.A.	Q	R	1, 2. 5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	N.A.	S/U(b),Q	SA	1
b. Inoperative	N.A.	S/U(b),Q	N.A.	1
c. Comparator	N.A.	S/U(b),Q	SA	1
7. REACTOR MODE SWITCH SHUTDOWN POSITION	N.A.	R	N.A.	3, 4

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

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

(c) Includes reactor manual control multiplexing system input.

* With THERMAL POWER $\geq$ 30% of RATED THERMAL POWER.

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

*** Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3.

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. The system meets the intent of IEEE-279 for nuclear power plant protection systems. 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 NRC Safety Evaluation Report (SER) (letter to T. A. Pickens from A. Thadani dated July 15, 1987. The bases for the trip settings of 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 safety analyses. 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.

BASES

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.

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

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. 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.

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 13 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 and the 3 second load center loading delay. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 13-second delay. It follows that checking the valve speeds and the 13-second time for emergency power establishment will establish the response time for the isolation functions.

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 an allowance for instrument drift specifically allocated 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. 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 maintenance outage times have been determined in accordance with NEDC-30936P, Parts 1 and 2, "Technical Speicification Improvement Methodology (with Demonstration for BWR ECCS

3/4.3.3 EMERGENCY CORE COOLING ACTUATION INSTRUMENTATION (Continued)

Actuation Instrumentation)," as approved by the NRC and documented in the SER (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)).

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 an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971, NEDO-24222, dated December 1979, and Section 15.8 of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a supplement to the reactor trip. During turbine trip and generator load rejection events, the EOC-RPT will reduce the likelihood of reactor vessel level decreasing to level 2. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. 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 sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch 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.

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 30% of RATED THERMAL POWER are annunciated in the control room.

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., 175 ms. Included in this time are: the response time of the sensor, the time allotted for breaker arc suppression, and the response time of the system logic.

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 an allowance for instrument drift specifically allocated 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. This instrumentation does not provide actuation of any of the emergency core cooling equipment.

Specified surveillance intervals and maintenance outage times have been specified in accordance with recommendations made by GE in their letter to the BWR Owner's Group dated August 7, 1989, SUBJECT: "Clarification of Technical Specification changes given in ECCS Actuation Instrumentation Analysis."

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 an allowance for instrument drift specifically allocated 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 and Section 3/4.3 Instrumentation. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

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

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 an allowance for instrument drift specifically allocated 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, and (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 10 CFR Part 50, Appendix A, General Design Criteria 19, 41, 60, 61, 63, and 64.

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 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE ODCM.

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 Part 50, Appendix A.

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions shall not be made without this flux level information available to the operator. When the intermediate range monitors are on scale, adequate information is available without the SRMs and they can be retracted.

3/4.3.7.7 TRAVERSING IN-CORE PROBE SYSTEM

The OPERABILITY of the traversing in-core probe system with the specified minimum complement of equipment ensures that the measurements obtained from use of this equipment accurately represent the spatial neutron flux distribution of the reactor core.

The TIP system OPERABILITY is demonstrated by normalizing all probes (i.e., detectors) prior to performing an LPRM calibration function. Monitoring core thermal limits may involve utilizing individual detectors to monitor selected areas of the reactor core, thus all detectors may not be required to be OPERABLE. The OPERABILITY of individual detectors to be used for monitoring is demonstrated by comparing the detector(s) output in the resultant heat balance calculation (P-1) with data obtained during a previous heat balance calculation (P-1).

3/4.3.7.8 CHLORINE AND TOXIC GAS DETECTION SYSTEMS

The OPERABILITY of the chlorine and toxic gas detection systems ensures that an accidental chlorine and/or toxic gas release will be detected promptly and the necessary protective actions will be automatically initiated for chlorine and manually initiated for toxic gas to provide protection for control room personnel. Upon detection of a high concentration of chlorine, the control room emergency ventilation system will automatically be placed in the chlorine isolation mode of operation to provide the required protection. Upon detection of a high concentration of toxic gas, the control room emergency ventilation system will manually be placed in the chlorine isolation mode of operation to provide the required protection. The detection systems required by this specification are consistent with the recommendations of Regulatory Guide 1.95 "Protection of Nuclear Power Plant Control Room Operators against an Accidental Chlorine Release," February 1975.

3/4.3.7.9 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the detection instrumentation ensures that both adequate warning capability is available for prompt detection of fires and that fire suppression systems, that are actuated by fire detectors, will discharge extinguishing agent in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

Fire detectors that are used to actuate fire suppression systems represent a more critically important component of a plant's fire protection program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of OPERABLE fire detectors must be greater.

The loss of detection capability for fire suppression systems, actuated by fire detectors, represents a significant degradation of fire protection for any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.7.10 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the loose-part detection system ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.7.11 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE ODCM.

effective January 2, 1991

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

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-353

LIMERICK GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 17
License No. NPF-85

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

   A. The application for amendment by Philadelphia Electric Company (the licensee) dated April 26, 1990, 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-85 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 17 , are hereby incorporated into this license. Philadelphia Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective fifteen (15) days after date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard J. Clark for

Charles L. Miller, Director
Project Directorate I-2
Division of Reactor Projects - I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 2, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 17

FACILITY OPERATING LICENSE NO. NPF-85

DOCKET NO. 50-353

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. Overleaf pages are provided to maintain document completeness.*

Remove	Insert
xix	xix
xx	xx*
3/4 3-1	3/4 3-1
3/4 3-2	3/4 3-2*
3/4 3-5	3/4 3-5
3/4 3-6	3/4 3-6*
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
3/4 3-9	3/4 3-9
3/4 3-10	3/4 3-10
3/4 3-15	3/4 3-15*
3/4 3-16	3/4 3-16
3/4 3-17	3/4 3-17
3/4 3-18	3/4 3-18*
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29*
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 3-32	3/4 3-32*
3/4 3-35	3/4 3-35
3/4 3-36	3/4 3-36
3/4 3-39	3/4 3-39*
3/4 3-40	3/4 3-40
3/4 3-41	3/4 3-41
3/4 3-42	3/4 3-42*

ATTACHMENT TO LICENSE AMENDMENT NO. 17

FACILITY OPERATING LICENSE NO. NPF-85

DOCKET NO. 50-353

Remove	Insert
3/4 3-53	3/4 3-53
3/4 3-54	3/4 3-54
3/4 3-55	3/4 3-55*
3/4 3-56	3/4 3-56
3/4 3-61	3/4 3-61
3/4 3-62	3/4 3-62*
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 3-4	B 3/4 3-4
B 3/4 3-5	B 3/4 3-5
B 3/4 3-6	B 3/4 3-6*

SECTION PAGE

INSTRUMENTATION (Continued)

BASES

SECTION PAGE

REACTOR COOLANT SYSTEM (Continued)

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within 12 hours. The provisions of Specification 3.0.4 are not applicable.

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

SURVEILLANCE REQUIREMENTS

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

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

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

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* An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 6 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

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

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors(b):			
a. Neutron Flux - High	2	3	1
	3, 4	3	2
	5(c)	3(d)	3
b. Inoperative	2	3	1
	3, 4	3	2
	5	3(d)	3
2. Average Power Range Monitor(e):			
a. Neutron Flux - Upscale, Setdown	2	2	1
	3	2	2
	5(c)(l)	2(d)	3
b. Neutron Flux - Upscale			
1) Flow Biased	1	2	4
2) High Flow Clamped	1	2	4
c. Inoperative	1, 2	2	1
	3	2	2
	5(c)(l)	2(d)	3
d. Downscale	1(g)	2	4
3. Reactor Vessel Steam Dome Pressure - High	1, 2(f)	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Main Steam Line Isolation Valve - Closure	1(g)	1/valve	4

RE_ OR PROTECTION SYSTEM INSTRUMEN_ ION

TABLE NOTATIONS

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

(b) This function shall be automatically bypassed when the reactor mode switch is in the Run position and the associated APRM is not downscale.

(c) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations performed per Specification 3.10.3.

(d) The noncoincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per Trip System is 4 APRMs, 6 IRMs and 2 SRMs.

(e) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

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

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

(h) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

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

(j) This function shall be automatically bypassed when turbine first stage pressure is equivalent to a THERMAL POWER of less than 30% of RATED THERMAL POWER.

(k) Also actuates the EOC-RPT system.

(l) Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3.

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*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	N.A.
b. Inoperative	N.A.
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale, Setdown	N.A.
b. Neutron Flux - Upscale	
1) Flow Biased	$\leq$0.09
2) High Flow Clamped	$\leq$0.09
c. Inoperative	N.A.
d. Downscale	N.A.
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ 0.55
4. Reactor Vessel Water Level - Low, Level 3	$\leq$ 1.05
5. Main Steam Line Isolation Valve - Closure	$\leq$ 0.06
6. Main Steam Line Radiation - High	N.A.
7. Drywell Pressure - High	N.A.
8. Scram Discharge Volume Water Level - High	
a. Level Transmitter	N.A.
b. Float Switch	N.A.
9. Turbine Stop Valve - Closure	$\leq$ 0.06
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\leq$ 0.08**
11. Reactor Mode Switch Shutdown Position	N.A.
12. Manual Scram	N.A.

*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.

**Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U,S(b)	S/U(c), W	R	2
	S	W(j)	R	3, 4, 5
b. Inoperative	N.A.	W(j)	N.A.	2, 3, 4, 5
2. Average Power Range Monitor[f]:				
a. Neutron Flux - Upscale, Setdown	S/U,S(b)	S/U(c), W	SA	2
	S	W(j)	SA	3, 5[k]
b. Neutron Flux - Upscale				
1) Flow Biased	S,D(g)	S/U(c), Q	W(d)(e), SA	1
2) High Flow Clamped	S	S/U(c), Q	W(d)(e), SA	1
c. Inoperative	N.A.	Q(j)	N.A.	1, 2, 3, 5[k]
d. Downscale	S	Q	SA	1
3. Reactor Vessel Steam Dome Pressure - High	S	Q	R	1, 2(h)
4. Reactor Vessel Water Level - Low, Level 3	S	Q	R	1, 2
5. Main Steam Line Isolation Valve - Closure	N.A.	Q	R	1
6. Main Steam Line Radiation - High	S	Q	R	1, 2(h)
7. Drywell Pressure - High	S	Q	R	1, 2
8. Scram Discharge Volume Water Level - High				
a. Level Transmitter	S	Q	R	1, 2, 5[i]
b. Float Switch	N.A.	Q	R	1, 2, 5[i]

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
9. Turbine Stop Valve - Closure	N.A.	Q	R	1
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	N.A.	Q	R	1
11. Reactor Mode Switch Shutdown Position	N.A.	R	N.A.	1, 2, 3, 4, 5
12. Manual Scram	N.A.	W	N.A.	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for a least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.
(c) Within 24 hours prior to startup, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.
(g) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing loop flow (APRM % flow). During the startup test program, data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon the conclusion of the startup test program.
(h) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.
(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
(j) If the RPS shorting links are required to be removed per Specification 3.9.2, they may be reinstalled for up to 2 hours for required surveillance. During this time, CORE ALTERATIONS shall be suspended, and no control rod shall be moved from its existing position.
(k) Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3.

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 channels(s) in the tripped condition would cause an isolation, the inoperable channel(s) shall be restored to OPERABLE status within:

   a) 2 hours for trip functions not common* to the Reactor Protection System (RPS) and/or Emergency Core Cooling System (ECCS) Actuation Instrumentation, or

   b) 6 hours for trip functions common* to RPS and/or ECCS Actuation Instrumentation.

   If this cannot be accomplished, the ACTION required by Table 3.3.2-1 for the affected trip function shall be taken, or the channel shall be placed in the tripped condition.

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) 1 hour for trip functions not common* to the RPS and/or ECCS Actuation Instrumentation,

   b) 12 hours for trip functions common* to RPS Instrumentation,

   c) 24 hours for trip functions common* to ECCS Actuation Instrumentation,

   and

   d) 12 hours for trip functions common* to RPS and ECCS Actuation Instrumentation.

The provisions of Specification 3.0.4 are not applicable.

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* Trip functions common to RPS and/or ECCS Actuation Instrumentation are shown in Table 4.3.2.1-1.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

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 1 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 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.

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

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL (a),(c)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
7. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level Low, Low - Level 2	B	2	1, 2, 3	25
b. Drywell Pressure - High	H	2	1, 2, 3	25
c.1. Refueling Area Unit 1 Ventilation Exhaust Duct Radiation - High	R	2	*#	25
2. Refueling Area Unit 2 Ventilation Exhaust Duct Radiation - High	R	2	*#	25
d. Reactor Enclosure Ventilation Exhaust Duct Radiation - High	S	2	1, 2, 3	25
e. Outside Atmosphere To Reactor Enclosure Δ Pressure - Low	U	1	1, 2, 3	25
f. Outside Atmosphere To Refueling Area Δ Pressure - Low	T	1	*	25
g. Reactor Enclosure Manual Initiation	NA	1	1, 2, 3	24
h. Refueling Area Manual Initiation	NA	1	*	25

ACTION STATEMENTS

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

ACTION 21 - 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 22 - Be in at least STARTUP within 6 hours.

ACTION 23 - In OPERATIONAL CONDITION 1 or 2, verify the affected system isolation valves are closed within 1 hour and declare the affected system inoperable. In OPERATIONAL CONDITION 3, be in at least COLD SHUTDOWN within 12 hours.

ACTION 24 - 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 or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 25 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

ACTION 26 - Close the affected system isolation valves within 1 hour.

TABLE NOTATIONS

* Required when (1) handling irradiated fuel in the refueling area secondary containment, or (2) during CORE ALTERATIONS, or (3) during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

** May be bypassed under administrative control, with all turbine stop valves closed.

\# During operation of the associated Unit 1 or Unit 2 ventilation exhaust system.

(a) See Specification 3.6.3, Table 3.6.3-1 for primary containment isolation valves which are actuated by these isolation signals.

(b) A channel may be placed in an inoperable status for up to:

a) 2 hours for trip functions not common the the Reactor Protection System (RPS) and/or Emergency Core Cooling System (ECCS) Actuation Instrumentation, or
b) 6 hours for trip functions common to RPS and/or ECCS Actuation Instrumentation

for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter. Trip functions common to RPS and/or ECCS Actuation Instrumentation are shown in Table 4.3.2.1-1. In addition, for the HPCI system and RCIC system isolation, provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is OPERABLE and all required actuation instrumentation for that valve is OPERABLE, one channel may be placed in an inoperable status for up to 8 hours for required surveillance without placing the channel or trip system in the tripped condition.

TABLE NOTATIONS

(c) Actuates secondary containment isolation valves shown in Table 3.6.5.2.1-1 and/or 3.6.5.2.2-1 and signals B, H, S, U, R and T also start the standby gas treatment system.

(d) RWCU system inlet outboard isolation valve closes on SLCS "B" initiation. RWCU system inlet inboard isolation valve closes on SLCS "A" or SLCS "C" initiation.

(e) Manual initiation isolates the steam supply line outboard isolation valve and only following manual or automatic initiation of the system.

(f) In the event of a loss of ventilation the temperature - high setpoint may be raised by 50°F for a period not to exceed 30 minutes to permit restoration of the ventilation flow without a spurious trip. During the 30 minute period, an operator, or other qualified member of the technical staff, shall observe the temperature indications continuously, so that, in the event of rapid increases in temperature, the main steam lines shall be manually isolated.

(g) Wide range accident monitor per Specification 3.3.7.5.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Low - Level 2	$\geq$ - 38 inches*	$\geq$ - 45 inches
2) Low, Low, Low - Level 1	$\geq$ - 129 inches*	$\geq$ - 136 inches
b. Main Steam Line Radiation - High	$\leq$ 3.0 x Full Power Background	$\leq$ 3.6 x Full Power Background
c. Main Steam Line Pressure - Low	$\geq$ 756 psig	$\geq$ 736 psig
d. Main Steam Line Flow - High	$\leq$ 108.7 psid	$\leq$ 111.7 psid
e. Condenser Vacuum - Low	10.5 psia	$\geq$ 10.1 psia/$\leq$ 10.9 psia
f. Outboard MSIV Room Temperature - High	$\leq$ 192°F	$\leq$ 200°F
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	$\leq$ 165°F	$\leq$ 175°F
h. Manual Initiation	N.A.	N.A.
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
a. Reactor Vessel Water Level Low - Level 3	$\geq$ 12.5 inches*	$\geq$ 11.0 inches
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	$\leq$ 75 psig	$\leq$ 95 psig
c. Manual Initiation	N.A.	N.A.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIR
1. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level##				
1) Low, Low, Level 2	S	Q	R	1, 2, 3
2) Low, Low, Low - Level 1	S	Q	R	1, 2, 3
b. Main Steam Line Radiation#### - High	S	Q	R	1, 2, 3
c. Main Steam Line Pressure - Low	S	M	R	1
d. Main Steam Line Flow - High	S	M	R	1, 2, 3
e. Condenser Vacuum - Low	S	M	R	1, 2**, 3**
f. Outboard MSIV Room Temperature - High	S	M	R	1, 2, 3
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	S	M	R	1, 2, 3
h. Manual Initiation	N.A.	R	N.A.	1, 2, 3
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level### Low - Level 3	S	Q	R	1, 2, 3
b. Reactor Vessel (RHR Cut-In Permissive) Pressure## - High	S	Q	R	1, 2, 3
c. Manual Initiation	N.A.	R	N.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 FOR WHICH SURVEILLANCE REQUIRED
3.	REACTOR WATER CLEANUP SYSTEM ISOLATION				
a.	RWCS Δ Flow - High	S	M	R	1, 2, 3
b.	RWCS Area Temperature - High	S	M	R	1, 2, 3
c.	RWCS Area Ventilation Δ Temperature - High	S	M	R	1, 2, 3
d.	SLCS Initiation	N.A.	R	N.A.	1, 2, 3
e.	Reactor Vessel Water Level## Low, Low, - Level 2	S	Q	R	1, 2, 3
f.	Manual Initiation	N.A.	R	N.A.	1, 2, 3
4.	HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a.	HPCI Steam Line Δ Pressure - High	S	M	R	1, 2, 3
b.	HPCI Steam Supply Pressure - Low	S	M	R	1, 2, 3
c.	HPCI Turbine Exhaust Diaphragm Pressure - High	S	M	R	1, 2, 3
d.	HPCI Equipment Room Temperature - High	S	M	R	1, 2, 3
e.	HPCI Equipment Room Δ Temperature - High	S	M	R	1, 2, 3
f.	HPCI Pipe Routing Area Temperature - High	S	M	R	1, 2, 3
g.	Manual Initiation	N.A.	R	N.A.	1, 2, 3
h.	HPCI Steam Line Δ Pressure Timer	N.A.	M	R	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 FOR WHICH SURVEILLANCE REQUIRED
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Δ Pressure - High	S	M	R	1, 2, 3
b. RCIC Steam Supply Pressure - Low	S	M	R	1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure - High	S	M	R	1, 2, 3
d. RCIC Equipment Room Temperature - High	S	M	R	1, 2, 3
e. RCIC Equipment Room Δ Temperature - High	S	M	R	1, 2, 3
f. RCIC Pipe Routing Area Temperature - High	S	M	R	1, 2, 3
g. Manual Initiation	N.A.	R	N.A.	1, 2, 3
h. RCIC Steam Line Δ Pressure Timer	N.A.	M	R	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 FOR WHIC[H] SURVEILLANCE REQUIRE[D]
6. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level##				
1) Low, Low - Level 2	S	Q	R	1, 2, 3
2) Low, Low, Low - Level 1	S	Q	R	1, 2, 3
b. Drywell Pressure### - High	S	Q	R	1, 2, 3
c. North Stack Effluent Radiation - High	S	Q	R	1, 2, 3
d. Deleted				
e. Reactor Enclosure Ventilation Exhaust Duct - Radiation - High	S	M	R	1, 2, 3
f. Outside Atmosphere to Reactor Enclosure Δ Pressure - Low	N.A.	M	Q	1, 2, 3
g. Deleted				
h. Drywell Pressure## - High/ Reactor Pressure## - Low	S	Q	R	1, 2, 3
i. Primary Containment Instrument Gas to Drywell Δ Pressure - Low	N.A.	M	Q	1, 2, 3
j. Manual Initiation	N.A.	R	N.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 FOR WHICH SURVEILLANCE REQUIRED
7. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level## Low, Low - Level 2	S	Q	R	1, 2, 3
b. Drywell Pressure### - High	S	Q	R	1, 2, 3
c. 1. Refueling Area Unit 1 Ventilation Exhaust Duct Radiation - High	S	M	R	*#
2. Refueling Area Unit 2 Ventilation Exhaust Duct Radiation - High	S	M	R	*#
d. Reactor Enclosure Ventilation Exhaust Duct Radiation - High	S	M	R	1, 2, 3
e. Outside Atmosphere To Reactor Enclosure Δ Pressure - Low	N.A.	M	Q	1, 2, 3
f. Outside Atmosphere To Refueling Area Δ Pressure - Low	N.A.	M	Q	*
g. Reactor Enclosure Manual Initiation	N.A.	R	N.A.	1, 2, 3
h. Refueling Area Manual Initiation	N.A.	R	N.A.	*

*Required when (1) handling irradiated fuel in the refueling area secondary containment, or (2) during CORE ALTERATIONS, or (3) during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

**When not administratively bypassed and/or when any turbine stop valve is open.

#During operation of the associated Unit 1 or Unit 2 ventilation exhaust system.

##These trip functions (1a, 2b, 3e, 6a, 6h, and 7a) are common to the ECCS actuation trip function.

###These trip functions (2a, 6b, and 7b) are common to the RPS and ECCS actuation trip functions.

####This trip function (1b) is common to the RPS trip function.

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 subsystem inoperable, restore the inoperable trip system to OPERABLE status within:

   1. 7 days, provided that the HPCI and RCIC systems are OPERABLE.

   2. 72 hours.

   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.

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TABLE NOTATIONS

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

(b) Also provides input to actuation logic for the associated emergency diesel generators.

(c) One trip system. Provides signal to HPCI pump suction valves only.

(d) On 1 out of 2 taken twice logic, provides a signal to trip the HPCI pump turbine only.

(e) The manual initiation push buttons start the respective core spray pump and diesel generator. The "A" and "B" logic manual push buttons also actuate an initiation permissive in the injection valve opening logic.

(f) A channel as used here is defined as the 127 bus relay for Item 1 and the 127, 127Y, and 127Z feeder relays with their associated time delay relays taken together for Item 2.

* When the system is required to be OPERABLE per Specification 3.5.2.

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

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

\#\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.

ACTION STATEMENTS

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 inoperable.

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

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

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within 24 hours.

ACTION 33 - 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 ECCS inoperable.

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. For one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or declare the HPCI system inoperable.

b. With more than one channel inoperable, declare the HPCI system inoperable.

ACTION 35 - 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 HPCI system inoperable.

ACTION 36 - With the number of OPERABLE channels less than the Total Number of Channels, declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2, as appropriate.

ACTION 37 - With the number of OPERABLE channels one less than the Total Number of Channels, place the inoperable channel in the tripped condition within 1 hour; operation may then continue until performance of the next required CHANNEL FUNCTIONAL TEST.

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

ECCS	RESPONSE TIME (Seconds)
1. CORE SPRAY SYSTEM	$\leq$ 27
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM	$\leq$ 40
3. AUTOMATIC DEPRESSURIZATION SYSTEM	N.A.
4. HIGH PRESSURE COOLANT INJECTION SYSTEM	$\leq$ 30
5. LOSS OF POWER	N.A.

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
1. CORE SPRAY SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	Q	R	1, 2, 3
c. Reactor Vessel Pressure - Low	S	Q	R	1, 2, 3, 4*, 5*
d. Manual Initiation	N.A.	R	N.A.	1, 2, 3, 4*, 5*
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	Q	R	1, 2, 3
c. Reactor Vessel Pressure - Low	S	Q	R	1, 2, 3
d. Injection Valve Differential Pressure - Low (Permissive)	S	Q	R	1, 2, 3, 4*, 5*
e. Manual Initiation	N.A.	R	N.A.	1, 2, 3, 4*, 5*
3. HIGH PRESSURE COOLANT INJECTION SYSTEM***				
a. Reactor Vessel Water Level - Low Low, Level 2	S	Q	R	1, 2, 3
b. Drywell Pressure - High	S	Q	R	1, 2, 3
c. Condensate Storage Tank Level - Low	S	Q	R	1, 2, 3
d. Suppression Pool Water Level - High	S	Q	R	1, 2, 3
e. Reactor Vessel Water Level - High, Level 8	S	Q	R	1, 2, 3
f. Manual Initiation	N.A.	R	N.A.	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
4. AUTOMATIC DEPRESSURIZATION SYSTEM#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	R	1, 2, 3
b. Drywell Pressure - High	S	Q	R	1, 2, 3
c. ADS Timer	N.A.	Q	Q	1, 2, 3
d. Core Spray Pump Discharge Pressure - High	S	Q	R	1, 2, 3
e. RHR LPCI Mode Pump Discharge Pressure - High	S	Q	R	1, 2, 3
f. Reactor Vessel Water Level - Low, Level 3	S	Q	R	1, 2, 3
g. Manual Initiation	N.A.	R	N.A.	1, 2, 3
h. ADS Drywell Pressure Bypass Timer	N.A.	Q	Q	1, 2, 3
5. LOSS OF POWER				
a. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)##	N.A.	R	N.A.	1, 2, 3, 4**, 5**
b. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	S	M	R	1, 2, 3, 4**, 5**

* When the system is required to be OPERABLE per Specification 3.5.2.

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

*** Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

\## Loss of Voltage Relay 127-11X is not field setable.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With an ATWS recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.1-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within 1 hour.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:

   1. If the inoperable channels consist of one reactor vessel water level channel and one reactor vessel pressure channel, place both inoperable channels in the tripped condition within 1 hour, or, if this action will initiate a pump trip, declare the trip system inoperable.

   2. If the inoperable channels include two reactor vessel water level channels or two reactor vessel pressure channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within 1 hour or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS.

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

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

REACTOR COR .SOLATION COOLING SYSTEM ACTUAT. INSTRUMENTATION

FUNCTIONAL UNITS		MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION*	ACTION
a.	Reactor Vessel Water Level - Low Low, Level 2	4#	50
b.	Reactor Vessel Water Level - High, Level 8	4#	51
c.	Condensate Storage Tank Water Level - Low	2**	52
d.	Manual Initiation	1/system***	53

*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 all other channels monitoring that parameter are OPERABLE.

**One trip system with one-out-of-two logic.

***One trip system with one channel.

#One trip system with one-out-of-two twice logic.

ACTION 50 - 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 RCIC system inoperable.

b. With more than one channel inoperable, declare the RCIC system inoperable.

ACTION 51 - With the number of OPERABLE channels less than required by the minimum OPERABLE channels per Trip System requirement, declare the RCIC system inoperable within 24 hours.

ACTION 52 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, place at least one inoperable channel in the tripped condition within 24 hours or declare the RCIC system inoperable.

ACTION 53 - With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the inoperable channel to OPERABLE status within 24 hours or declare the RCIC system inoperable.

TABLE 3.3.5-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNITS	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Water Level - Low Low, Level 2	$\geq$-38 inches*	$\geq$-45 inches
b. Reactor Vessel Water Level - High, Level 8	$\leq$ 54 inches	$\leq$ 60 inches
c. Condensate Storage Tank Level - Low	$\geq$ 135.8** inches	$\geq$ 132.3 inches
d. Manual Initiation	N.A.	N.A.

*See Bases Figure B 3/4.3-1.

**Corresponds to 2.3 feet indicated.

REACTOR CORE ISOLATION SYSTEM ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level - Low Low, Level 2	S	Q	R
b. Reactor Vessel Water Level - High, Level 8	S	Q	R
c. Condensate Storage Tank Level - Low	S	Q	R
d. Manual Initiation	N.A.	R	N.A.

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 BLOCK MONITOR				
a. Upscale	N.A.	S/U[(b)(c)],Q[(c)]	SA	1*
b. Inoperative	N.A.	S/U[(b)(c)],Q[(c)]	N.A.	1*
c. Downscale	N.A.	S/U[(b)(c)],Q[(c)]	SA	1*
2. APRM				
a. Flow Biased Neutron Flux - Upscale	N.A.	S/U[(b)],Q	SA	1
b. Inoperative	N.A.	S/U[(b)],Q	N.A.	1, 2, 5***
c. Downscale	N.A.	S/U[(b)],Q	SA	1
d. Neutron Flux - Upscale, Startup	N.A.	S/U[(b)],Q	SA	2, 5***
3. SOURCE RANGE MONITORS				
a. Detector not full in	N.A.	S/U[(b)],W	N.A.	2, 5
b. Upscale	N.A.	S/U[(b)],W	SA	2, 5
c. Inoperative	N.A.	S/U[(b)],W	N.A.	2, 5
d. Downscale	N.A.	S/U[(b)],W	SA	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	N.A.	S/U[(b)],W	N.A.	2, 5
b. Upscale	N.A.	S/U[(b)],W	SA	2, 5
c. Inoperative	N.A.	S/U[(b)],W	N.A.	2, 5
d. Downscale	N.A.	S/U[(b)],W	SA	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	N.A.	Q	R	1, 2, 5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	N.A.	S/U[(b)],Q	SA	1
b. Inoperative	N.A.	S/U[(b)],Q	N.A.	1
c. Comparator	N.A.	S/U[(b)],Q	SA	1
7. REACTOR MODE SWITCH SHUTDOWN POSITION	N.A.	R	N.A.	3, 4

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

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

(c) Includes reactor manual control multiplexing system input.

* With THERMAL POWER ≥ 30% of RATED THERMAL POWER.

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

*** Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3.

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. The system meets the intent of IEEE-279 for nuclear power plant protection systems. 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 NRC Safety Evaluation Report (SER) (letter to T. A. Pickens from A. Thadani dated July 15, 1987. The bases for the trip settings of 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 safety analyses. 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.

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

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. 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.

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 13 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 and the 3 second load center loading delay. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 13-second delay. It follows that checking the valve speeds and the 13-second time for emergency power establishment will establish the response time for the isolation functions.

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 an allowance for instrument drift specifically allocated 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. 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 maintenance outage times have been determined in accordance with NEDC-30936P, Parts 1 and 2, "Technical Specification Improvement Methodology (with Demonstration for BWR ECCS Actuation Instrumentation)" as approved by the NRC and documented in the SER (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)).

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 an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971, NEDO-24222, dated December 1979, and Section 15.8 of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a supplement to the reactor trip. During turbine trip and generator load rejection events, the EOC-RPT will reduce the likelihood of reactor vessel level decreasing to level 2. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. 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 sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch 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.

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 30% of RATED THERMAL POWER are annunciated in the control room.

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., 175 ms. Included in this time are: the response time of the sensor, the time allotted for breaker arc suppression, and the response time of the system logic.

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 an allowance for instrument drift specifically allocated 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. This instrumentation does not provide actuation of any of the emergency core cooling equipment.

Specified surveillance intervals and maintenance outage times have been specified in accordance with recommendations made by GE in their letter to the BWR Owner's Group dated August 7, 1989, SUBJECT: "Clarification of Technical Specification changes given in ECCS Actuation Instrumentation Analysis."

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 an allowance for instrument drift specifically allocated 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 and Section 3/4.3 Instrumentation. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

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

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 an allowance for instrument drift specifically allocated 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, and (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 10 CFR Part 50, Appendix A, General Design Criteria 19, 41, 60, 61, 63, and 64.

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 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE ODCM.

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 Part 50, Appendix A. The Unit 1 RHR transfer switches are included only due to their potential impact on the RHRSW system, which is common to both units.

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions shall not be made without this flux level information available to the operator. When the intermediate range monitors are on scale, adequate information is available without the SRMs and they can be retracted.

3/4.3.7.7 TRAVERSING IN-CORE PROBE SYSTEM

The OPERABILITY of the traversing in-core probe system with the specified minimum complement of equipment ensures that the measurements obtained from use of this equipment accurately represent the spatial neutron flux distribution of the reactor core.

The TIP system OPERABILITY is demonstrated by normalizing all probes (i.e., detectors) prior to performing an LPRM calibration function. Monitoring core thermal limits may involve utilizing individual detectors to monitor selected areas of the reactor core, thus all detectors may not be required to be OPERABLE. The OPERABILITY of individual detectors to be used for monitoring is demonstrated by comparing the detector(s) output in the resultant heat balance calculation (P-1) with data obtained during a previous heat balance calculation (P-1).

3/4.3.7.8 CHLORINE AND TOXIC GAS DETECTION SYSTEMS

The OPERABILITY of the chlorine and toxic gas detection systems ensures that an accidental chlorine and/or toxic gas release will be detected promptly and the necessary protective actions will be automatically initiated for chlorine and manually initiated for toxic gas to provide protection for control room personnel. Upon detection of a high concentration of chlorine, the control room emergency ventilation system will automatically be placed in the chlorine isolation mode of operation to provide the required protection. Upon detection of a high concentration of toxic gas, the control room emergency ventilation system will manually be placed in the chlorine isolation mode of operation to provide the required protection. The detection systems required by this specification are consistent with the recommendations of Regulatory Guide 1.95 "Protection of Nuclear Power Plant Control Room Operators against an Accidental Chlorine Release," February 1975.

3/4.3.7.9 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the detection instrumentation ensures that both adequate warning capability is available for prompt detection of fires and that fire suppression systems, that are actuated by fire detectors, will discharge extinguishing agent in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

Fire detectors that are used to actuate fire suppression systems represent a more critically important component of a plant's fire protection program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of OPERABLE fire detectors must be greater.

The loss of detection capability for fire suppression systems, actuated by fire detectors, represents a significant degradation of fire protection for any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.7.10 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the loose-part detection system ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.7.11 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE ODCM.

Effective January 2, 1991

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 53 AND 17 TO FACILITY OPERATING

LICENSE NOS. NPF-39 AND NPF-85

LIMERICK GENERATING STATION, UNITS 1 AND 2

DOCKET NOS. 50-353

1.0 INTRODUCTION

By letter dated April 26, 1990, Philadelphia Electric Company (the licensee) requested an amendment to Facility Operating License Nos. NPF-39 and NPF-85 for the Limerick Generating Station, Units 1 and 2 (LGS-1&2). These proposed amendments would revise the Technical Specifications (TS) for LGS-1&2 to extend the surveillance test intervals (STIs) and allowable out-of-service times (AOTs) for instrumentation supporting the Reactor Protection System (RPS) and Emergency Core Cooling System (ECCS), including instrumentation common to the Control Rod Block Function (CRBF), the Reactor Core Isolation Cooling (RCIC) system, and the isolation instrumentation common to RPS and/or ECCS.

These changes are based upon the BWR Owners Group (BWROG) Topical Reports NEDC-30851P, "Technical Specification Improvement Analysis for BWR Reactor Protection System," May 1985, which provided a safety analysis for increased surveillance test intervals and allowed out-of-service times for RPS instrumentation on a generic basis; NEDC-30851P, supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," June 1986, which provided a safety analysis for extension of on-line test intervals for control rod block instrumentation on a generic basis; and NEDC-30851P, Supplement 2, "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," July 1986, which provided a safety analysis for extension of Surveillance Test Intervals and allowed outage times for isolation instrumentation common to RPS and ECCS instrumentation. The NRC staff reviewed NEDC-30851P and its supplements, and issued Safety Evaluation's (SE) on July 15, 1987, September 22, 1988, and Janauary 6, 1989, respectively, approving the reports and providing model TS changes. The staff's original SE was incorporated into an approved version of the topical report, NEDC-30851P-A, issued April 4, 1988.

The BWROG Topical Report NEDC-30936P, " Technical Specification Improvement Methodology (with demonstration for BWR ECCS actuation Instrumentation) Parts 1 and 2," dated November 1985, provided the justification for increased surveillance test intervals and allowed out-of-service time for ECCS instrumentation on a generic basis. On December 9, 1988, the NRC staff issued a Safety Evaluation on "Review of BWR Owners Group Report NEDC-30936P (Part 2) on Justification for Extending On-Line Test Intervals and Allowable Out-of-Service Times for BWR Emergency Core Cooling System Instrumentation."

As discussed on page 3 of Attachment 1 of the licensee's application, the NRC staff's evaluation did not cover the RCIC instrumentation or the End-of-Cycle Recirculation Pump Trip (EOC-RPT) instrumentation. To resolve this omission, by letter dated February 19, 1991, the BWR Owners Group submitted two topical reports: GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications" and GENE-770-06-2, Addendum to "Bases for Changes to Surveillance Test Intervals and Allowed Out of Service Times for Selected Instrumentation Technical Specifications". The latter provided the results of additional analyses to support changes to the STI and AOT for RCIC instrumentation. By letter dated September 13, 1991, the NRC staff issued a Safety Evaluation Report (SER) and supporting Technical Evaluation Report (TER) on the topical report GENE-770-06-2. Enclosure 2 to the letter provided model TSs. The staff concluded that GENE-770-06-2 provided an acceptable basis for extending STEs and AOTs for RCIC actuation instrumentation.

As noted above, GENE-770-06-1 addresses STI and AOT changes such as the EOC-RPT which were originally proposed in the Improved BWR Technical Specifications but which were not specifically covered by a previous NRC Safety Evaluation. The NRC staff has not completed its review of this topical report. By agreement with the licensee, rather than holdup processing of the subject application any longer, the TS changes related to EOC-RPT (pages 3/4 3-46, 3/4 3-48, 3/4 3-51 and part of B 3/4 3-3) that were included in the subject application of April 26, 1990 are not included in these amendments. If the proposed TS changes on EOC-RPT are found to be acceptable when the staff completes its review of the topical report, they will be included in a future, separate amendment.

The staff's generic SER stated that plant specific application of the generic results would require comparison of the plant specific design with the generic design to show that NEDC-30851P-A and NEDC-30936P-A are applicable and that any increase in instrument drift due to the extended STI is properly accounted for in the setpoint calculation methodology.

The licensee's submittal, dated April 26, 1990, included the General Electric Company (GE) Reports MDE-93-0485-1, dated October 1987, which compared LGS-1&2 RPS design with that used in NEDC-30851P-A, and RE-019, dated December 1986, which compared LGS1&2 ECCS design with that used in NEDC-30936P-A. The submittal also provided PECo's response to the plant specific conditions required to be met by the staff's generic SER, and included supplemental data regarding the drift of RPS and ECCS instrumentation.

2.0 EVALUATION

The NRC staff has reviewed the licensee's April 26, 1990, submittal. The proposed TS changes reflect those standard TS revisions contained in NEDC-30851P-A and NEDC-30936P-A which, based upon probabilistic analyses, justify the identified time extensions by reducing the potential for: (1) unnecessary plant scrams; (2) excessive equipment test cycles; and (3) diversion of personnel and resources to conduct unnecessary testing.

The licensee has extended the generic analysis completed by the BWR Owners Group to LGS-1&2 by having General Electric Company complete the required plant specific analysis. As stated in the NRC's SERs for Licensing Topical Reports, three conditions must be addressed to justify the applicability of the generic analysis to individual plants when specific facility TS are considered for revision.

1. Confirm the applicability of the generic analysis to the specific facility.

   The licensee's submittal GE report MDE-93-0485-1, "Technical Specification Improvement Analysis for the Reactor Protection System for Limerick Generating Station Units 1&2," dated October 1987, identified that the RPS configuration for LGS 1&2 has several differences compared to the RPS configuration in the generic evaluation. These differences and the assessment of their effects on the RPS failure frequency were evaluated. The licensee concluded that these differences would not significantly affect the improvement in plant safety due to the changes in the technical specifications based on the generic analysis. The generic analysis in NEDC-30851P-A is applicable to the LGS 1&2 RPS. The licensee's submittal GE Report RE-19," Technical Specification Improvement Analysis for Emergency Core Cooling System Actuation Instrumentation for Limerick Generating Station, Units 1&2," dated December 1986 identified that the ECCS configuration for LGS is similar to the ECCS configuration in the generic evaluation, with only one significant difference-i.e. the generic model has four emergency service water loop while LGS only has two loops. This difference between LGS and the generic model has been modeled by envelope Case 4A of NEDC-30936P-A which shows that the proposed changes to ECCS Actuation instrumentation Technical Specifications would meet the 4% acceptance criterion in NEDC-30936P-A. Therefore, the licensee concluded that the generic basis in NEDC-30936P-A is applicable to the LGS 1&2 ECCS actuation instrumentation. The staff has reviewed the comparison documentation in the licensee's submittals and concur with the licensee's conclusion that the generic analyses are applicable to the LGS Units 1&2.

2. Demonstrate that the drift characteristics for RPS and ECCS channel instrumentation are bounded by the assumptions used in NEDC-30851P-A and NEDC-30936P-A when the functional test interval is extended from monthly to quarterly.

   The additional time interval between tests resulting from the changes described in NEDC-30851P-A, NEDC-30936P-A and requested in this submittal, is already factored into the instrument setpoint calculations for the affected instruments. As stated in the Bases to LGS-1&2 TS 3/4.3.1 and TS 3/4.3.3, the difference between each RPS and ECCS instrument trip setpoint and the allowable value is equal to or greater than the drift allowance assumed for each trip in the plant safety analyses. The setpoint calculations for both the RPS and the ECCS assume an eighteen month calibration interval and the drift based upon vendor supplied values associated with that interval with no credit taken for the currently specified 31 day functional test. This assumption in the setpoint calculations, therefore, bounds any drift which could be expected over the 92 day functional test interval proposed. The licensee has also provided data from surveillance tests at LGS-1 for a representative sample of RPS and ECCS instrumentation and demonstrated that instrumentation drift characteristics are bounded for the proposed surveillance test interval. Accordingly, revised instrument setpoints or allowable values are not required to accommodate the longer test intervals requested. Since similar instruments are used at LGS-2, the data from LGS -1 are applicable to both units.

3. Confirm that the differences between the parts of the RPS (and ECCS) that perform the trip functions in the plant and those of the base case plant were included in the specific analysis done using the procedures of Appendix K to NEDC-30851P-A.

In the licensee's submittals, GE reports MDE-93-0485-1, and RE-019, which utilize the procedures of the Licensing Topical Report NEDC-30851P-A Appendix K to identify and evaluate the differences between the parts of RPS and ECCS that perform the trip functions at LGS-1&2 and those of the base case plant. The results indicate that while the RPS configuration for LGS-1&2 has several differences compared to the configuration in the base case, the differences and their impact do not significantly affect the applicability of the TS changes developed by the generic efforts of Licensing Topical Report NEDC-30851P-A or NEDC-30936P-A.

The staff has reviewed the plant specific report for LGS and has verified that the differences between the LGS-1&2 RPS and ECCS at the time the analysis was made and the generic RPS and ECCS were included in the plant specific analysis. Since the plant specific analyses were done (October 1987, and December 1986,) no RPS or ECCS modifications have been made which would invalidate the conclusions of these reports. Therefore, the conclusions reached in NEDC-30851P-A and NEDC-30936P-A apply to LGS-1&2 and the plant-specific changes contained in this request are bounded by both the generic analyses and the NRC's generic safety evaluations.

The acceptability of the revised TSs has been addressed in the various staff safety evaluations on the topical reports. The assessment of the LCO action requirements in the revised Section 3.3.1 is addressed below.

For 3.3.1a, with the number of operable channels less than required in only one trip system i.e. any number of Functional Units having only one inoperable channel in each Function Unit the entire RPS scram capability remains intact, assuming no additional single failure. The condition which allows continued operation for 12 hours has been evaluated and the reliability of the system shown to be acceptable in NEDC-30851P-A, March 1988. Within 12 hours the inoperable channels and/or trip system must be placed in the tripped condition. This action restores the RPS capability to accommodate a single failure and allows operation to continue with no further restrictions. If the inoperable channel(s) and/or trip system is not placed in the tripped condition within the allowed time, then the ACTION required by Table 3.3.1-1 must be taken to place the plant in a condition where the function limits are not required to be OPERABLE.

For 3.3.1b, with the number of operable channels less than required in both trip systems, such that they may have two or more channels inoperable in any Functional Unit, the Reactor Protection System may not be capable of performing its intended function (e.g., a loss of scram function may exist). In this condition, during the 12 hour allowance to place the inoperable channels and/or trip system in the tripped condition, if a valid trip signal was received a failure to automatically scram could result. In order to reduce the probability of this occurrence, the action for this condition requires that steps be taken to ensure each required Functional Unit maintains trip capability for each Trip System within 1 hour. This time period allows the operator time to evaluate, to repair or trip the channels. This time period is reasonable considering the diversity of sensors available to provide trip signals, and the low probability of an event requiring the initiation of a scram. This time period is also consistent with the current Technical Specifications which address this condition. In addition, when one channel in each trip system is INOPERABLE, an allowance of 6 hours is provided in order to restore an additional level of RPS Reliability. The requirement to ensure for each Functional Unit the required channel(s) in one Trip System, or the Trip System are OPERABLE or in the tripped condition limits the time the RPS scram logic for any Functional Unit would not accommodate a single failure in either Trip System. The 6 hour time period is considered acceptable based on the remaining capability to trip, the diversity available to provide trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse functions, and the low probability allowance, each Functional Unit will have all required channels OPERABLE or in TRIP in one Trip System. This provides a similar level of RPS reliability as found in Action a, above, and evaluated in NEDC-30851P-A to be acceptable for a 12 hour allowed outage time.

Based on the staff's evaluation of the licensee's submittal, the staff finds that LGS-1&2 has met the plant specific conditions needed to apply the results of GE's Topical Report NEDC-30851P to the Limerick Generating Station, Units 1&2. The proposed changes to the TSs are acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (55 FR 21975). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

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

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