

April 26, 1994

Docket Nos. 50-352  
and 50-353

Mr. George A. Hunger, Jr.  
Director-Licensing, MC 52A-5  
Philadelphia Electric Company  
Nuclear Group Headquarters  
Correspondence Control Desk  
P.O. Box No. 195  
Wayne, Pennsylvania 19087-0195

Dear Mr. Hunger:

**SUBJECT: EXTEND SURVEILLANCE TEST INTERVALS AND ALLOWED OUTAGE TIMES FOR  
SELECTED ACTUATION INSTRUMENTATION, LIMERICK GENERATING STATION,  
UNITS 1 AND 2 (TAC NOS. M86507 AND M86508)**

The Commission has issued the enclosed Amendment No. 70 to Facility Operating License No. NPF-39 and Amendment No. 33 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 response to your application dated May 6, 1993, as supplemented by letter dated April 18, 1994.

These amendments provide an extension of surveillance test intervals and allowed outage times (AOTs) for selected actuation instrumentation TS and make editorial changes. The approved amendments will minimize testing and remove restrictive AOTs that could potentially degrade the overall plant safety and availability.

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

Sincerely,  
/s/

Frank Rinaldi, Project Manager  
Project Directorate I-2  
Division of Reactor Projects - I/II  
Office of Nuclear Reactor Regulation

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Enclosures:

1. Amendment No. 70 to License No. NPF-39  
Amendment No. 33 to License No. NPF-85
2. Safety Evaluation

**NRC FILE REVIEW COPY**

cc w/enclosures:

See next page

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Docket File	MO'Brien(2)	CGrimes, 11E21
NRC & Local PDRs	FRinaldi/JShea	JWermiel
PDI-2 Reading	OGC	ACRS(10)
SVarga	DHagan, 3206	OPA
JCalvo	GHill(4), P1-22	OC/LFDCB
CMiller	EWenzinger, RGN-I	CAnderson, RGN-I
*Previous Concurrence		

*CP1*

OFC	: PDI-2/LA	: PDI-2/PM	: *HICB	: *OGC	: PDI-2/D	:	:
NAME	: MC'Brien	: FRinaldi	: rb: JWermiel	: RBachmann	: CMiller	:	:
DATE	: 4/25/94	: 4/25/94	: 08/23/93	: 03/23/94	: 4/26/94	:	:

*JFD/11*

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UNITED STATES  
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001  
April 26, 1994

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Sincerely,

A handwritten signature in cursive script, appearing to read "Frank Rinaldi".

Frank Rinaldi, Project Manager  
Project Directorate I-2  
Division of Reactor Projects - I/II  
Office of Nuclear Reactor Regulation

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See next page

Mr. George A. Hunger, Jr.  
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Limerick Generating Station,  
Units 1 & 2

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

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-352

LIMERICK GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 70  
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 May 6, 1993, as supplemented by letter dated April 18, 1994, 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.

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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. 70 , 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 as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



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

Attachment:  
Changes to the  
Technical Specifications

Date of Issuance: April 26, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 70

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

3/4 1-5  
3/4 1-6

3/4 1-5  
3/4 1-6\*

3/4 3-41  
3/4 3-42

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3/4 3-115

3/4 4-7  
3/4 4-8

3/4 4-7  
3/4 4-8\*

ATTACHMENT TO LICENSE AMENDMENT NO. 70

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.\*

<u>Remove</u>	<u>Insert</u>
3/4 6-13	3/4 6-13*
3/4 6-14	3/4 6-14
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
B 3/4 3-7	B 3/4 3-7
B 3/4 3-8	B 3/4 3-8*

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

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

- a. The scram discharge volume drain and vent valves OPERABLE, when control rods are scram tested from a normal control rod configuration of less than or equal to 50% ROD DENSITY at least once per 18 months, by verifying that the drain and vent valves:
  1. Close within 30 seconds after receipt of a signal for control rods to scram, and
  2. Open when the scram signal is reset.
- b. Proper level sensor response by performance of a CHANNEL FUNCTIONAL TEST of the scram discharge volume scram and control rod block level instrumentation at least once per 92 days.

## REACTIVITY CONTROL SYSTEMS

### CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

#### LIMITING CONDITION FOR OPERATION

---

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 5, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

#### ACTION:

- a. With the maximum scram insertion time of one or more control rods exceeding 7 seconds:
  1. Declare the control rod(s) with the slow insertion time inoperable, and
  2. Perform the Surveillance Requirements of Specification 4.1.3.2c. at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of 7.0 seconds.

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

- b. The provisions of Specification 3.0.4 are not applicable.

#### SURVEILLANCE REQUIREMENTS

---

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

- a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days.
- b. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and
- c. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

TABLE 4.3 .1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>CHANNEL CALIBRATION</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>
<b>4. <u>AUTOMATIC DEPRESSURIZATION SYSTEM</u></b>				
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
<b>5. <u>LOSS OF POWER</u></b>				
a. 4.16 kV Emergency Bus Under-voltage (Loss of Voltage)##	N.A.	R	N.A.	1, 2, 3, 4**, 5**
b. 4.16 kV Emergency Bus Under-voltage (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 settable.

LIMERICK - UNIT 1

3/4 3-41

Amendment No. 53

Effective December 17, 1991

## 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 24 hours.
- 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 24 hours, 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.

TABLE 3.3.4.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

<u>TRIP FUNCTION</u>	<u>MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM *</u>
1. Reactor Vessel Water Level - Low Low, Level 2	2
2. Reactor Vessel Pressure - High	2

---

\* One channel may be placed in an inoperable status for up to 6 hours for required surveillance provided the other channel is OPERABLE.

TABLE 3.3.4.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

<u>TRIP FUNCTION</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
1. Reactor Vessel, Water Level - Low Low, Level 2	$\geq -38$ inches*	$\geq -45$ inches
2. Reactor Vessel Pressure - High	$\leq 1093$ psig	$\leq 1108$ psig

---

\*See Bases Figure B3/4 3-1.

TABLE 4.3.4.1-1

ATWS RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>CHANNEL CALIBRATION</u>
1. Reactor Vessel Water Level - Low Low, Level 2	S	Q	R
2. Reactor Vessel Pressure - High	S	Q	R

## INSTRUMENTATION

### END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

#### LIMITING CONDITION FOR OPERATION

---

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER.

#### ACTION:

- a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel 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 12 hours.
- 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 turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within 12 hours.
  2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system inoperable.
- d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or take the ACTION required by Specification 3.2.3.
- e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or take the ACTION required by Specification 3.2.3.

## INSTRUMENTATION

### SURVEILLANCE REQUIREMENTS

---

4.3.4.2.1 Each end-of-cycle recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.2.1-1.

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

4.3.4.2.3 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME of each trip function shown in Table 3.3.4.2-3 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least the logic of one type of channel input, turbine control valve fast closure or turbine stop valve closure, such that both types of channel inputs are tested at least once per 36 months. The measured time shall be added to the most recent breaker arc suppression time and the resulting END-OF-CYCLE-RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be verified to be within its limit.

4.3.4.2.4 The time interval necessary for breaker arc suppression from energization of the recirculation pump circuit breaker trip coil shall be measured at least once per 60 months.

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

<u>TRIP FUNCTION</u>	<u>MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM*</u>
1. Turbine Stop Valve - Closure	2**
2. Turbine Control Valve-Fast Closure	2**

---

\* A trip system may be placed in an inoperable status for up to 6 hours for required surveillance provided that the other trip system is OPERABLE.

\*\* This function shall be automatically bypassed when turbine first stage pressure is equivalent to THERMAL POWER LESS than 30% of RATED THERMAL POWER.

TABLE 4.3.4.2.1-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>CHANNEL CALIBRATION</u>
1. Turbine Stop Valve-Closure	Q*	R
2. Turbine Control Valve-Fast Closure	Q*	R

\* Including trip system logic testing.

## INSTRUMENTATION

### 3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

#### LIMITING CONDITION FOR OPERATION

---

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

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

#### ACTION:

- a. With a RCIC system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.5-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
- b. With one or more RCIC system actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.5-1.

#### SURVEILLANCE REQUIREMENTS

---

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

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

## INSTRUMENTATION

### 3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

#### LIMITING CONDITION FOR OPERATION

---

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

APPLICABILITY: As shown in Table 3.3.6-1.

#### ACTION:

- a. With a control rod block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
- b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

#### SURVEILLANCE REQUIREMENTS

---

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

\*A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition, provided at least one other operable channel in the same trip system is monitoring that parameter.

TABLE 3.3.6-1  
CONTROL ROD BLOCK INSTRUMENTATION

<u>TRIP FUNCTION</u>	<u>MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION</u>	<u>APPLICABLE OPERATIONAL CONDITIONS</u>	<u>ACTION</u>
1. <u>ROD BLOCK MONITOR</u> <sup>(a)</sup>			
a. Upscale	2	1*	60
b. Inoperative	2	1*	60
c. Downscale	2	1*	60
2. <u>APRM</u>			
a. Flow Biased Neutron Flux - Upscale	4	1	61
b. Inoperative	4	1, 2, 5 <sup>(f)</sup>	61
c. Downscale	4	1	61
d. Neutron Flux - Upscale, Startup	4	2, 5 <sup>(f)</sup>	61
3. <u>SOURCE RANGE MONITORS</u> ***			
a. Detector not full in <sup>(b)</sup>	3	2	61
	2	5	61
b. Upscale <sup>(c)</sup>	3	2	61
	2	5	61
c. Inoperative <sup>(c)</sup>	3	2	61
	2	5	61
d. Downscale <sup>(d)</sup>	3	2	61
	2	5	61
4. <u>INTERMEDIATE RANGE MONITORS</u>			
a. Detector not full in	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative <sup>(e)</sup>	6	2, 5	61
d. Downscale <sup>(e)</sup>	6	2, 5	61
5. <u>SCRAM DISCHARGE VOLUME</u>			
a. Water Level-High	2	1, 2, 5**	62
6. <u>REACTOR COOLANT SYSTEM RECIRCULATION FLOW</u>			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. Comparator	2	1	62
7. <u>REACTOR MODE SWITCH SHUTDOWN POSITION</u>	2	3, 4	63

TABLE 3.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

ACTION STATEMENTS

- ACTION 60 - Declare the RBM inoperable and take the ACTION required by Specification 3.1.4.3.
- ACTION 61 - With the number of OPERABLE Channels:
- a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 12 hours or place the inoperable channel in the tripped condition.
  - b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.
- ACTION 62 - 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 12 hours.
- ACTION 63 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, initiate a rod block.

NOTES

- \* For OPERATIONAL CONDITION of Specification 3.1.4.3.
- \*\* With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
- \*\*\* These channels are not required when sixteen or fewer fuel assemblies, adjacent to the SRMs, are in the core.
- (a) The RBM shall be automatically bypassed when a peripheral control rod is selected or the reference APRM channel indicates less than 30% of RATED THERMAL POWER.
  - (b) This function shall be automatically bypassed if detector count rate is > 100 cps or the IRM channels are on range 3 or higher.
  - (c) This function is automatically bypassed when the associated IRM channels are on range 8 or higher.
  - (d) This function is automatically bypassed when the IRM channels are on range 3 or higher.
  - (e) This function is automatically bypassed when the IRM channels are on range 1.
  - (f) Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3.

**TABLE 3.3.6-2  
CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS**

<u>TRIP FUNCTION</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
<b>1. <u>ROD BLOCK MONITOR</u></b>		
a. <u>Upscale<sup>(a)</sup></u>		
1) Low Trip Setpoint (LTSP)	*	*
2) Intermediate Trip Setpoint (ITSP)	*	*
3) High Trip Setpoint (HTSP)	*	*
b. Inoperative	N/A	N/A
c. <u>Downscale (DTSP)</u>	*	*
d. <u>Power Range Setpoint<sup>(b)</sup></u>		
1) Low Power Setpoint (LPSP)	23% RATED THERMAL POWER	26% RATED THERMAL POWER
2) Intermediate Power Setpoint (IPSP)	58% RATED THERMAL POWER	61% RATED THERMAL POWER
3) High Power Setpoint (HPSP)	78% RATED THERMAL POWER	81% RATED THERMAL POWER
<b>2. <u>APRM</u></b>		
a. <u>Flow Biased Neutron Flux - Upscale</u>		
1) During two recirculation loop operation	$\leq 0.66 W + 59\%$	$\leq 0.66 W + 63\%$
2) During single recirculation loop operation	$\leq 0.66 W + 54\%$	$\leq 0.66 W + 58\%$
b. Inoperative	N.A.	N.A.
c. <u>Downscale</u>	$\geq 4\%$ of RATED THERMAL POWER	$\geq 3\%$ of RATED THERMAL POWER
d. <u>Neutron Flux - Upscale, Startup</u>	$\leq 12\%$ of RATED THERMAL POWER	$\leq 14\%$ of RATED THERMAL POWER
<b>3. <u>SOURCE RANGE MONITORS</u></b>		
a. Detector not full in	N.A.	N.A.
b. <u>Upscale</u>	$\leq 1 \times 10^5$ cps	$\leq 1.6 \times 10^5$ cps
c. Inoperative	N.A.	N.A.
d. <u>Downscale</u>	$\geq 3$ cps**	$\geq 1.8$ cps**
<b>4. <u>INTERMEDIATE RANGE MONITORS</u></b>		
a. Detector not full in	N.A.	N.A.
b. <u>Upscale</u>	$\leq 108/125$ divisions of full scale	$\leq 110/125$ divisions of full scale
c. Inoperative	N.A.	N.A.
d. <u>Downscale</u>	$\geq 5/125$ divisions of full scale	$\geq 3/125$ divisions of full scale
<b>5. <u>SCRAM DISCHARGE VOLUME</u></b>		
a. <u>Water Level-High</u>		
a. Float Switch	$\leq 257' 5 \frac{9}{16}"$ elevation***	$\leq 257' 7 \frac{9}{16}"$ elevation

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Amendment No. 7, 19, 30, 37, 66

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

TABLE NOTATIONS

\*When irradiated fuel is being handled in the secondary containment.

(a) With fuel in the spent fuel storage pool.

(b) Alarm only.

ACTION STATEMENTS

- ACTION 70 - With one monitor inoperable, restore the inoperable monitor to the OPERABLE status within 7 days or, within the next 6 hours, initiate and maintain operation of the control room emergency filtration system in the radiation isolation mode of operation.
- With two or more of the monitors inoperable, within one hour, initiate and maintain operation of the control room emergency filtration system in the radiation mode of operation.
- ACTION 71 - With one of the required monitor inoperable, assure a portable continuous monitor with the same alarm setpoint is OPERABLE in the vicinity of the installed monitor during any fuel movement. If no fuel movement is being made, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.
- ACTION 72 - With the required monitor inoperable, obtain and analyze at least one grab sample of the monitored parameter at least once per 24 hours.
- ACTION 73 - With the required monitor inoperable, assure a portable alarming monitor is OPERABLE in the vicinity of the installed monitor or perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

TABLE 4.3.7.1-1RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>INSTRUMENTATION</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>CHANNEL CALIBRATION</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE IS REQUIRED</u>
1. Main Control Room Normal Fresh Air Supply Radiation Monitor	S	Q	R	1, 2, 3, 5 and *
2. Area Monitors				
a. Criticality Monitors				
1) Spent Fuel Storage Pool	S	M	R	(a)
b. Control Room Direct Radiation Monitor	S	M	R	At All Times
3. Reactor Enclosure Cooling Water Radiation Monitor	S	M	R(b)	At All Times

## INSTRUMENTATION

### TRAVERSING IN-CORE PROBE SYSTEM

#### LIMITING CONDITION FOR OPERATION

---

3.3.7.7 The traversing in-core probe system shall be OPERABLE with:

- a. Five movable detectors, drives and readout equipment to map the core, and
- b. Indexing equipment to allow all five detectors to be calibrated in a common location.

APPLICABILITY: When the traversing in-core probe is used for:

- a. Recalibration of the LPRM detectors, and
- b.\* Monitoring the APLHGR, LHGR, MCPR, or MFLPD.

#### ACTION:

With the traversing in-core probe system inoperable, suspend use of the system for the above applicable monitoring or calibration functions. The provisions of Specification 3.0.3 are not applicable.

#### SURVEILLANCE REQUIREMENTS

---

4.3.7.7 The traversing in-core probe system shall be demonstrated OPERABLE by normalizing each of the above required detector outputs within 72 hours prior to use for the LPRM calibration function.

---

\*Only the detector(s) in the required measurement location(s) are required to be OPERABLE.

## INSTRUMENTATION

### CHLORINE DETECTION SYSTEM

#### LIMITING CONDITION FOR OPERATION

---

3.3.7.8.1 Two independent chlorine detection system subsystems shall be OPERABLE with their alarm and trip setpoints adjusted to actuate at a chlorine concentration of less than or equal to 0.5 ppm

APPLICABILITY: ALL OPERATIONAL CONDITIONS.

#### ACTION:

- a. With one chlorine detection subsystem inoperable, restore the inoperable detection system to OPERABLE status within 7 days or, within the next 6 hours, initiate and maintain operation of at least one control room emergency filtration system subsystem in the chlorine isolation mode of operation.
- b. With both chlorine detection subsystems inoperable, within 1 hour initiate and maintain operation of at least one control room emergency filtration system subsystem in the chlorine isolation mode of operation.

#### SURVEILLANCE REQUIREMENTS

---

4.3.7.8.1 Each of the above required chlorine detection system subsystems shall be demonstrated OPERABLE by performance of a:

- a. CHANNEL CHECK at least once per 12 hours,
- b. CHANNEL FUNCTIONAL TEST at least once per 92 days, and
- c. CHANNEL CALIBRATION at least once per 18 months.

INSTRUMENTATION

SURVEILLANCE REQUIREMENTS (Continued)

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- b. At least once per 31 days by:
  - 1. Cycling each of the following valves through at least one complete cycle from the running position:
    - a) For the overspeed protection control system;
      - 1) Four high pressure turbine control valves
    - b) For the electrical overspeed trip system and the mechanical overspeed trip system;
      - 1) Four high pressure turbine control valves
- c. At least once per 18 months by performance of a CHANNEL CALIBRATION of the turbine overspeed protection instrumentation.
- d. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of all valve seats, disks and stems and verifying no unacceptable flaws or excessive corrosion. If unacceptable flaws or excessive corrosion are found, all other valves of that type shall be inspected.

## INSTRUMENTATION

### 3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

#### LIMITING CONDITION FOR OPERATION

3.3.9 The feedwater/main turbine trip system actuation instrumentation channels shown in the Table 3.3.9-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.9-2.

APPLICABILITY: As shown in Table 3.3.9-1.

#### ACTION:

- a. With a feedwater/main turbine trip system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.9-2, declare the channel inoperable and either place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value, or declare the associated system inoperable.
- b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least STARTUP within the next 6 hours.
- c. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

#### SURVEILLANCE REQUIREMENTS

4.3.9.1 Each feedwater/main turbine trip system actuation instrumentation channel shall be demonstrated OPERABLE\* by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.9.1-1.

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

\*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.

TABLE 4.3.9.1-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION  
SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>CHANNEL CALIBRATION</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>
1. Reactor Vessel Water Level-High, Level 8	D	Q	R	1

## REACTOR COOLANT SYSTEM

### 3/4.4.2 SAFETY/RELIEF VALVES

#### LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of at least 11 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:\*

- 4 safety/relief valves @ 1130 psig  $\pm$  1%
- 5 safety/relief valves @ 1140 psig  $\pm$  1%
- 5 safety/relief valves @ 1150 psig  $\pm$  1%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

#### ACTION:

- a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
- b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 105°F, close the stuck open safety/relief valve(s); if unable to close the stuck open valve(s) within 2 minutes or if suppression pool average water temperature is 110°F or greater, place the reactor mode switch in the Shutdown position.
- c. With one or more safety/relief valve acoustic monitors inoperable, restore the inoperable acoustic monitors to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

#### SURVEILLANCE REQUIREMENTS

4.4.2.1 The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be 0.20 of the full open noise level## by performance of a:

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

4.4.2.2 At least 1/2 of the safety relief valves shall be removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations at least once per 24 months, and they shall be rotated such that all 14 safety relief valves are removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations at least once per 54 months.

- \* The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.
- \*\* The provisions of Specification 4.0.4 are not applicable provided the Surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.
- # Up to 2 inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until the next refueling.
- ## Initial setting shall be in accordance with the manufacturer's recommendation. Adjustment to the valve full open noise level shall be accomplished during the startup test program.

## REACTOR COOLANT SYSTEM

### 3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

#### LEAKAGE DETECTION SYSTEMS

#### LIMITING CONDITION FOR OPERATION

---

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

- a. The primary containment atmosphere gaseous radioactivity monitoring system,
- b. The drywell floor drain sump and drywell equipment drain tank flow monitoring system,
- c. The drywell unit coolers condensate flow rate monitoring system, and
- d. The primary containment pressure and temperature monitoring system.

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

#### ACTION:

With only three of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous radioactive monitoring system, primary containment pressure and temperature monitoring system and/or the drywell unit coolers condensate flow rate monitoring system is inoperable; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

#### SURVEILLANCE REQUIREMENTS

---

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

- a. Primary containment atmosphere gaseous radioactivity monitoring systems-performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.
- b. The primary containment pressure shall be monitored at least once per 12 hours and the primary containment temperature shall be monitored at least once per 24 hours.
- c. Drywell floor drain sump and Drywell equipment drain tank flow monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per 18 months.
- d. Drywell unit coolers condensate flow rate monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

---

\*The primary containment atmosphere gaseous radioactivity monitor is not required to be OPERABLE until OPERATIONAL CONDITION 2.

## CONTAINMENT SYSTEMS

### LIMITING CONDITION FOR OPERATION (Continued)

#### ACTION: (Continued)

3. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.
- c. With only one suppression chamber water level indicator OPERABLE and/or with less than eight suppression pool water temperature indicators, one in each of the eight locations OPERABLE, restore the inoperable indicator(s) to OPERABLE status within 7 days or verify suppression chamber water level and/or temperature to be within the limits at least once per 12 hours.
- d. With no suppression chamber water level indicators OPERABLE and/or with less than seven suppression pool water temperature indicators covering at least seven locations OPERABLE, restore at least one water level indicator and at least seven water temperature indicators to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- e. With the drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 200°F.

### SURVEILLANCE REQUIREMENTS

- 4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:
  - a. By verifying the suppression chamber water volume to be within the limits at least once per 24 hours.
  - b. At least once per 24 hours by verifying the suppression chamber average water temperature to be less than or equal to 95°F, except:
    1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature less than or equal to 105°F.
    2. At least once per hour when suppression chamber average water temperature is greater than or equal to 95°F, by verifying:
      - a) Suppression chamber average water temperature to be less than or equal to 110°F, and
      - b) THERMAL POWER to be less than or equal to 1% of RATED THERMAL POWER 12 hours after suppression chamber average water temperature has exceeded 95°F for more than 24 hours.
    3. At least once per 30 minutes following a scram with suppression chamber average water temperature greater than or equal to 95°F, by verifying suppression chamber average water temperature less than or equal to 120°F.

## CONTAINMENT SYSTEMS

### SURVEILLANCE REQUIREMENTS (Continued)

- c. By verifying at least 8 suppression pool water temperature indicators in at least 8 locations, OPERABLE by performance of a:
1. CHANNEL CHECK at least once per 24 hours.
  2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
  3. CHANNEL CALIBRATION at least once per 18 months,

with the temperature alarm setpoint for:

1. High water temperature:
    - a) First setpoint  $\leq 95^{\circ}\text{F}$
    - b) Second setpoint  $\leq 105^{\circ}\text{F}$
    - c) Third setpoint  $\leq 110^{\circ}\text{F}$
    - d) Fourth setpoint  $\leq 120^{\circ}\text{F}$
- d. By verifying at least two suppression chamber water level indicators OPERABLE by performance of a:
1. CHANNEL CHECK at least once per 24 hours,
  2. CHANNEL FUNCTIONAL TEST at least once per 92 days, and
  3. CHANNEL CALIBRATION at least once per 18 months,

with the water level alarm setpoint for high water level  $\leq 24'1\text{-}1/2''$

- e. Drywell-to-suppression chamber bypass leak tests shall be conducted at 40 +/- 10 month intervals to coincide with the ILRT at an initial differential pressure of 4 psi and verifying that the  $A/\sqrt{k}$  calculated from the measured leakage is within the specified limit. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 24 months until two consecutive tests meet the specified limit, at which time the test schedule may be resumed.
- f. By conducting a leakage test on the drywell-to-suppression chamber vacuum breakers at a differential pressure of at least 4.0 psi and verifying that the total leakage area  $A/\sqrt{k}$  contributed by all vacuum breakers is less than or equal to 24% of the specified limit and the leakage area for an individual set of vacuum breakers is less than or equal to 12% of the specified limit. The vacuum breaker leakage test shall be conducted during each refueling outage for which the drywell-to-suppression chamber bypass leak test in Specification 4.6.2.1.d is not conducted.

## INSTRUMENTATION

### BASES

#### 3/4.3.3 EMERGENCY CORE COOLING ACTUATION INSTRUMENTATION (Continued)

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.

## INSTRUMENTATION

### BASES

#### 3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION (Continued)

Specified surveillance intervals and maintenance outage times have been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R.D. Binz, IV, from C.E. Rossi dated July 21, 1992).

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.

## INSTRUMENTATION

### BASES

#### 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.

The specified surveillance interval for the Main Control Room Normal Fresh Air Supply Radiation Monitor has been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R.D. Binz, IV, from C.E. Rossi dated July 21, 1992).

##### 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.

## INSTRUMENTATION

### BASES

#### 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.

Specified surveillance intervals and maintenance outage times have been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R.D. Binz, IV, from C.E. Rossi dated July 21, 1992).

#### 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.

## INSTRUMENTATION

### BASES

#### FIRE DETECTION INSTRUMENTATION (Continued)

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.

The surveillance requirements for demonstrating the OPERABILITY of the fire detectors are based on the recommendations of NFPA 72E - 1990 Edition.

#### 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.

#### 3/4.3.7.12 OFFGAS MONITORING INSTRUMENTATION

This instrumentation includes provisions for monitoring the concentrations of potentially explosive gas mixtures and noble gases in the off-gas system.

#### 3/4.3.8. TURBINE OVERSPEED PROTECTION SYSTEM

This specification is provided to ensure that the turbine overspeed protection system instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

#### 3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

The feedwater/main turbine trip system actuation instrumentation is provided to initiate action of the feedwater system/main turbine trip system in the event of failure of feedwater controller under maximum demand.





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

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-353

LIMERICK GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 33  
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 May 6, 1993, as supplemented by letter dated April 18, 1994, 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.

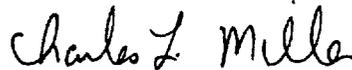
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-39 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 33, 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 as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



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

Attachment:  
Changes to the  
Technical Specifications

Date of Issuance: April 26, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 33

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

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ATTACHMENT TO LICENSE AMENDMENT NO. 33

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

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## REACTIVITY CONTROL SYSTEMS

### SURVEILLANCE REQUIREMENTS (Continued)

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

- a. The scram discharge volume drain and vent valves OPERABLE, when control rods are scram tested from a normal control rod configuration of less than or equal to 50% ROD DENSITY at least once per 18 months, by verifying that the drain and vent valves:
  1. Close within 30 seconds after receipt of a signal for control rods to scram, and
  2. Open when the scram signal is reset.
- b. Proper level sensor response by performance of a CHANNEL FUNCTIONAL TEST of the scram discharge volume scram and control rod block level instrumentation at least once per 92 days.

## REACTIVITY CONTROL SYSTEMS

### CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

#### LIMITING CONDITION FOR OPERATION

---

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3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 5, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

#### ACTION:

- a. With the maximum scram insertion time of one or more control rods exceeding 7 seconds:
  1. Declare the control rod(s) with the slow insertion time inoperable, and
  2. Perform the Surveillance Requirements of Specification 4.1.3.2c. at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of 7.0 seconds.

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

- b. The provisions of Specification 3.0.4 are not applicable.

#### SURVEILLANCE REQUIREMENTS

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4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

- a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days.
- b. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and
- c. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

TABLE 4.3. -1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>CHANNEL CALIBRATION</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>
<b>4. <u>AUTOMATIC DEPRESSURIZATION SYSTEM</u></b>				
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
<b>5. <u>LOSS OF POWER</u></b>				
a. 4.16 kV Emergency Bus Under-voltage (Loss of Voltage) <b>##</b>	N.A.	R	N.A.	1, 2, 3, 4**, 5**
b. 4.16 kV Emergency Bus Under-voltage (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 settable.

LIMERICK - UNIT 2

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Amendment No. 17

effective December 17, 1991

## 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 24 hours.
- 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 24 hours, 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

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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.

**TABLE 3.3.4.1-1**

**ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION**

<b><u>TRIP FUNCTION</u></b>	<b><u>MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM *</u></b>
1. Reactor Vessel Water Level - Low Low, Level 2	2
2. Reactor Vessel Pressure - High	2

---

\* One channel may be placed in an inoperable status for up to 6 hours for required surveillance provided the other channel is OPERABLE.

TABLE 3.3.4.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

<u>TRIP FUNCTION</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
1. Reactor Vessel, Water Level - Low Low, Level 2	$\geq -38$ inches*	$\geq -45$ inches
2. Reactor Vessel Pressure - High	$\leq 1093$ psig	$\leq 1108$ psig

---

\*See Bases Figure B3/4.3-1.

TABLE 4.3.4.1-1

ATWS RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>CHANNEL CALIBRATION</u>
1. Reactor Vessel Water Level - Low Low, Level 2	S	Q	R
2. Reactor Vessel Pressure - High	S	Q	R

## INSTRUMENTATION

### END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

#### LIMITING CONDITION FOR OPERATION

---

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER.

#### ACTION:

- a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel 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 12 hours.
- 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 turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within 12 hours.
  2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system inoperable.
- d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or take the ACTION required by Specification 3.2.3.
- e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or take the ACTION required by Specification 3.2.3.

## INSTRUMENTATION

### SURVEILLANCE REQUIREMENTS

---

4.3.4.2.1 Each end-of-cycle recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.2.1-1.

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

4.3.4.2.3 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME of each trip function shown in Table 3.3.4.2-3 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least the logic of one type of channel input, turbine control valve fast closure or turbine stop valve closure, such that both types of channel inputs are tested at least once per 36 months. The measured time shall be added to the most recent breaker arc suppression time and the resulting END-OF-CYCLE-RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be verified to be within its limit.

4.3.4.2.4 The time interval necessary for breaker arc suppression from energization of the recirculation pump circuit breaker trip coil shall be measured at least once per 60 months.

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

<u>TRIP FUNCTION</u>	<u>MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM*</u>
1. Turbine Stop Valve - Closure	2**
2. Turbine Control Valve-Fast Closure	2**

---

\* A trip system may be placed in an inoperable status for up to 6 hours for required surveillance provided that the other trip system is OPERABLE.

\*\* This function shall be automatically bypassed when turbine first stage pressure is equivalent to THERMAL POWER LESS than 30% of RATED THERMAL POWER.

TABLE 4.3.4.2.1-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>CHANNEL CALIBRATION</u>
1. Turbine Stop Valve-Closure	Q*	R
2. Turbine Control Valve-Fast Closure	Q*	R

\* Including trip system logic testing.

## INSTRUMENTATION

### 3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

#### LIMITING CONDITION FOR OPERATION

---

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

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

#### ACTION:

- a. With a RCIC system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.5-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
- b. With one or more RCIC system actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.5-1.

#### SURVEILLANCE REQUIREMENTS

---

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

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

## INSTRUMENTATION

### 3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

#### LIMITING CONDITION FOR OPERATION

---

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

APPLICABILITY: As shown in Table 3.3.6-1.

#### ACTION:

- a. With a control rod block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
- b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per trip Function requirement, take the ACTION required by Table 3.3.6-1.

#### SURVEILLANCE REQUIREMENTS

---

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

\* A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition, provided at least one other operable channel in the same trip system is monitoring that parameter.

TABLE 3.3.6-1  
CONTROL ROD BLOCK INSTRUMENTATION

<u>TRIP FUNCTION</u>	<u>MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION</u>	<u>APPLICABLE OPERATIONAL CONDITIONS</u>	<u>ACTION</u>
1. <u>ROD BLOCK MONITOR</u> <sup>(a)</sup>			
a. Upscale	2	1*	60
b. Inoperative	2	1*	60
c. Downscale	2	1*	60
2. <u>APRM</u>			
a. Flow Biased Neutron Flux - Upscale	4	1	61
b. Inoperative	4	1, 2, 5 <sup>(f)</sup>	61
c. Downscale	4	1	61
d. Neutron Flux - Upscale, Startup	4	2, 5 <sup>(f)</sup>	61
3. <u>SOURCE RANGE MONITORS</u> ***			
a. Detector not full in <sup>(b)</sup>	3	2	61
	2	5	61
b. Upscale <sup>(c)</sup>	3	2	61
	2	5	61
c. Inoperative <sup>(c)</sup>	3	2	61
	2	5	61
d. Downscale <sup>(d)</sup>	3	2	61
	2	5	61
4. <u>INTERMEDIATE RANGE MONITORS</u>			
a. Detector not full in	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative <sup>(e)</sup>	6	2, 5	61
d. Downscale <sup>(e)</sup>	6	2, 5	61
5. <u>SCRAM DISCHARGE VOLUME</u>			
a. Water Level-High	2	1, 2, 5**	62
6. <u>REACTOR COOLANT SYSTEM RECIRCULATION FLOW</u>			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. Comparator	2	1	62
7. <u>REACTOR MODE SWITCH SHUTDOWN POSITION</u>	2	3, 4	63

LIMERICK - UNIT 2

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Amendment No. 7

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TABLE 3.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

ACTION STATEMENTS

- ACTION 60 - Declare the RBM inoperable and take the ACTION required by Specification 3.1.4.3.
- ACTION 61 - With the number of OPERABLE Channels:
- a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 12 hours or place the inoperable channel in the tripped condition.
  - b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.
- ACTION 62 - 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 12 hours.
- ACTION 63 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, initiate a rod block.

NOTES

- \* 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.
  - \*\*\* These channels are not required when sixteen or fewer fuel assemblies, adjacent to the SRMs, are in the core.
- (a) The RBM shall be automatically bypassed when a peripheral control rod is selected or the reference APRM channel indicates less than 30% of RATED THERMAL POWER.
  - (b) This function shall be automatically bypassed if detector count rate is  $> 100$  cps or the IRM channels are on range 3 or higher.
  - (c) This function is automatically bypassed when the associated IRM channels are on range 8 or higher.
  - (d) This function is automatically bypassed when the IRM channels are on range 3 or higher.
  - (e) This function is automatically bypassed when the IRM channels are on range 1.
  - (f) Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3.

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

TABLE NOTATIONS

\*When irradiated fuel is being handled in the secondary containment.

(a) With fuel in the spent fuel storage pool.

(b) Alarm only.

ACTION STATEMENTS

- ACTION 70 -** With one monitor inoperable, restore the inoperable monitor to the OPERABLE status within 7 days or, within the next 6 hours, initiate and maintain operation of the control room emergency filtration system in the radiation isolation mode of operation.
- With two or more of the monitors inoperable, within one hour, initiate and maintain operation of the control room emergency filtration system in the radiation mode of operation.
- ACTION 71 -** With one of the required monitor inoperable, assure a portable continuous monitor with the same alarm setpoint is OPERABLE in the vicinity of the installed monitor during any fuel movement. If no fuel movement is being made, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.
- ACTION 72 -** With the required monitor inoperable, obtain and analyze at least one grab sample of the monitored parameter at least once per 24 hours.
- ACTION 73 -** With the required monitor inoperable, assure a portable alarming monitor is OPERABLE in the vicinity of the installed monitor or perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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<u>INSTRUMENTATION</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>CHANNEL CALIBRATION</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE IS REQUIRED</u>
1. Main Control Room Normal Fresh Air Supply Radiation Monitor	S	Q	R	1, 2, 3, 5 and *
2. Area Monitors				
a. Criticality Monitors				
1) Spent Fuel Storage Pool	S	M	R	(a)
b. Control Room Direct Radiation Monitor	S	M	R	At All Times
3. Reactor Enclosure Cooling Water Radiation Monitor	S	M	R(b)	At All Times

## INSTRUMENTATION

### TRAVERSING IN-CORE PROBE SYSTEM

#### LIMITING CONDITION FOR OPERATION

---

3.3.7.7 The traversing in-core probe system shall be OPERABLE with:

- a. Five movable detectors, drives and readout equipment to map the core, and
- b. Indexing equipment to allow all five detectors to be calibrated in a common location.

APPLICABILITY: When the traversing in-core probe is used for:

- a. Recalibration of the LPRM detectors, and
- b.\* Monitoring the APLHGR, LHGR, MCPR, or MFLPD.

#### ACTION:

With the traversing in-core probe system inoperable, suspend use of the system for the above applicable monitoring or calibration functions. The provisions of Specification 3.0.3 are not applicable.

#### SURVEILLANCE REQUIREMENTS

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4.3.7.7 The traversing in-core probe system shall be demonstrated OPERABLE by normalizing each of the above required detector outputs within 72 hours prior to use for the LPRM calibration function.

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\*Only the detector(s) in the required measurement location(s) are required to be OPERABLE.

## INSTRUMENTATION

### CHLORINE DETECTION SYSTEM

#### LIMITING CONDITION FOR OPERATION

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3.3.7.8.1 Two independent chlorine detection system subsystems shall be OPERABLE with their alarm and trip setpoints adjusted to actuate at a chlorine concentration of less than or equal to 0.5 ppm

APPLICABILITY: ALL OPERATIONAL CONDITIONS.

#### ACTION:

- a. With one chlorine detection subsystem inoperable, restore the inoperable detection system to OPERABLE status within 7 days or, within the next 6 hours, initiate and maintain operation of at least one control room emergency filtration system subsystem in the chlorine isolation mode of operation.
- b. With both chlorine detection subsystems inoperable, within 1 hour initiate and maintain operation of at least one control room emergency filtration system subsystem in the chlorine isolation mode of operation.

#### SURVEILLANCE REQUIREMENTS

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4.3.7.8.1 Each of the above required chlorine detection system subsystems shall be demonstrated OPERABLE by performance of a:

- a. CHANNEL CHECK at least once per 12 hours,
- b. CHANNEL FUNCTIONAL TEST at least once per 92 days, and
- c. CHANNEL CALIBRATION at least once per 18 months.

## INSTRUMENTATION

### SURVEILLANCE REQUIREMENTS (Continued)

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- b. At least once per 31 days by:
  - 1. Cycling each of the following valves through at least one complete cycle from the running position:
    - a) For the overspeed protection control system;
      - 1) Four high pressure turbine control valves
    - b) For the electrical overspeed trip system and the mechanical overspeed trip system;
      - 1) Four high pressure turbine control valves
- c. At least once per 18 months by performance of a CHANNEL CALIBRATION of the turbine overspeed protection instrumentation.
- d. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of all valve seats, disks and stems and verifying no unacceptable flaws or excessive corrosion. If unacceptable flaws or excessive corrosion are found, all other valves of that type shall be inspected.

## INSTRUMENTATION

### 3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

#### LIMITING CONDITION FOR OPERATION

3.3.9 The feedwater/main turbine trip system actuation instrumentation channels shown in the Table 3.3.9-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.9-2.

APPLICABILITY: As shown in Table 3.3.9-1.

#### ACTION:

- a. With a feedwater/main turbine trip system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.9-2, declare the channel inoperable and either place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value, or declare the associated system inoperable.
- b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least STARTUP within the next 6 hours.
- c. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

#### SURVEILLANCE REQUIREMENTS

4.3.9.1 Each feedwater/main turbine trip system actuation instrumentation channel shall be demonstrated OPERABLE\* by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.9.1-1.

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

\* 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.

TABLE 4.3.9.1-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION  
SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>CHANNEL CALIBRATION</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>
1. Reactor Vessel Water Level-High, Level 8	D	Q	R	1

## REACTOR COOLANT SYSTEM

### 3/4.4.2 SAFETY/RELIEF VALVES

#### LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of at least 11 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:\*

- 4 safety/relief valves @ 1130 psig  $\pm$  1%
- 5 safety/relief valves @ 1140 psig  $\pm$  1%
- 5 safety/relief valves @ 1150 psig  $\pm$  1%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

#### ACTION:

- a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
- b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 105°F, close the stuck open safety/relief valve(s); if unable to close the stuck open valve(s) within 2 minutes or if suppression pool average water temperature is 110°F or greater, place the reactor mode switch in the Shutdown position.
- c. With one or more safety/relief valve acoustic monitors inoperable, restore the inoperable acoustic monitors to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

#### SURVEILLANCE REQUIREMENTS

4.4.2.1 The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be 0.20 of the full open noise level## by performance of a:

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

4.4.2.2 At least 1/2 of the safety relief valves shall be removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations at least once per 24 months, and they shall be rotated such that all 14 safety relief valves are removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations at least once per 54 months.

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

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

# Up to 2 inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until the next refueling.

## Initial setting shall be in accordance with the manufacturer's recommendation. Adjustment to the valve full open noise level shall be accomplished during the startup test program.

## REACTOR COOLANT SYSTEM

### 3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

#### LEAKAGE DETECTION SYSTEMS

#### LIMITING CONDITION FOR OPERATION

---

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

- a. The primary containment atmosphere gaseous radioactivity monitoring system,
- b. The drywell floor drain sump and drywell equipment drain tank flow monitoring system,
- c. The drywell unit coolers condensate flow rate monitoring system, and
- d. The primary containment pressure and temperature monitoring system.

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

#### ACTION:

With only three of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous radioactive monitoring system, primary containment pressure and temperature monitoring system and/or the drywell unit coolers condensate flow rate monitoring system is inoperable; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

#### SURVEILLANCE REQUIREMENTS

---

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

- a. Primary containment atmosphere gaseous radioactivity monitoring systems-performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.
- b. The primary containment pressure shall be monitored at least once per 12 hours and the primary containment temperature shall be monitored at least once per 24 hours.
- c. Drywell floor drain sump and Drywell equipment drain tank flow monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per 18 months.
- d. Drywell unit coolers condensate flow rate monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

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\*The primary containment atmosphere gaseous radioactivity monitor is not required to be OPERABLE until OPERATIONAL CONDITION 2.

## CONTAINMENT SYSTEMS

### LIMITING CONDITION FOR OPERATION (Continued)

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#### ACTION: (Continued)

3. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.
- c. With only one suppression chamber water level indicator OPERABLE and/or with less than eight suppression pool water temperature indicators, one in each of the eight locations OPERABLE, restore the inoperable indicator(s) to OPERABLE status within 7 days or verify suppression chamber water level and/or temperature to be within the limits at least once per 12 hours.
- d. With no suppression chamber water level indicators OPERABLE and/or with less than seven suppression pool water temperature indicators covering at least seven locations OPERABLE, restore at least one water level indicator and at least seven water temperature indicators to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- e. With the drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 200°F.

### SURVEILLANCE REQUIREMENTS

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#### 4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

- a. By verifying the suppression chamber water volume to be within the limits at least once per 24 hours.
- b. At least once per 24 hours by verifying the suppression chamber average water temperature to be less than or equal to 95°F, except:
  1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature less than or equal to 105°F.
  2. At least once per hour when suppression chamber average water temperature is greater than or equal to 95°F, by verifying:
    - a) Suppression chamber average water temperature to be less than or equal to 110°F, and
    - b) THERMAL POWER to be less than or equal to 1% of RATED THERMAL POWER 12 hours after suppression chamber average water temperature has exceeded 95°F for more than 24 hours.
  3. At least once per 30 minutes following a scram with suppression chamber average water temperature greater than or equal to 95°F, by verifying suppression chamber average water temperature less than or equal to 120°F.

## CONTAINMENT SYSTEMS

### SURVEILLANCE REQUIREMENTS (Continued)

c. By verifying at least 8 suppression pool water temperature indicators in at least 8 locations, OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours.
2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
3. CHANNEL CALIBRATION at least once per 18 months,

with the temperature alarm setpoint for:

1. High water temperature:
  - a) First setpoint  $\leq 95^{\circ}\text{F}$
  - b) Second setpoint  $\leq 105^{\circ}\text{F}$
  - c) Third setpoint  $\leq 110^{\circ}\text{F}$
  - d) Fourth setpoint  $\leq 120^{\circ}\text{F}$

d. By verifying at least two suppression chamber water level indicators OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,
2. CHANNEL FUNCTIONAL TEST at least once per 92 days, and
3. CHANNEL CALIBRATION at least once per 18 months,

with the water level alarm setpoint for high water level  $\leq 24'1\text{-}1/2''$

e. Drywell-to-suppression chamber bypass leak tests shall be conducted at 40 +/- 10 month intervals to coincide with the ILRT at an initial differential pressure of 4 psi and verifying that the  $A/\sqrt{k}$  calculated from the measured leakage is within the specified limit. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 24 months until two consecutive tests meet the specified limit, at which time the test schedule may be resumed.

f. By conducting a leakage test on the drywell-to-suppression chamber vacuum breakers at a differential pressure of at least 4.0 psi and verifying that the total leakage area  $A/\sqrt{k}$  contributed by all vacuum breakers is less than or equal to 24% of the specified limit and the leakage area for an individual set of vacuum breakers is less than or equal to 12% of the specified limit. The vacuum breaker leakage test shall be conducted during each refueling outage for which the drywell-to-suppression chamber bypass leak test in Specification 4.6.2.1.d is not conducted.

## INSTRUMENTATION

### BASES

#### 3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION (Continued)

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.

**INSTRUMENTATION**  
**BASES**

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Specified surveillance intervals and maintenance outage times have been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R.D. Binz, IV, from C.E. Rossi dated July 21, 1992).

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 time 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.

## INSTRUMENTATION

### BASES

#### 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.

The specified surveillance interval for the Main Control Room Normal Fresh Air Supply Radiation Monitor has been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R.D. Binz, IV, from C.E. Rossi dated July 21, 1992).

##### 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.

## INSTRUMENTATION BASES

### 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 ventilating 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.

Specified surveillance intervals and maintenance outage times have been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R.D. Binz, IV, from C.E. Rossi dated July 21, 1992).

### 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.

## INSTRUMENTATION

### BASES

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#### 3/4.3.7.9 FIRE DETECTION INSTRUMENTATION (Continued)

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.

The surveillance requirements for demonstrating the OPERABILITY of the fire detectors are based on the recommendations of NFPA 72E - 1990 Edition.

#### 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.

#### 3/4.3.7.12 OFFGAS MONITORING INSTRUMENTATION

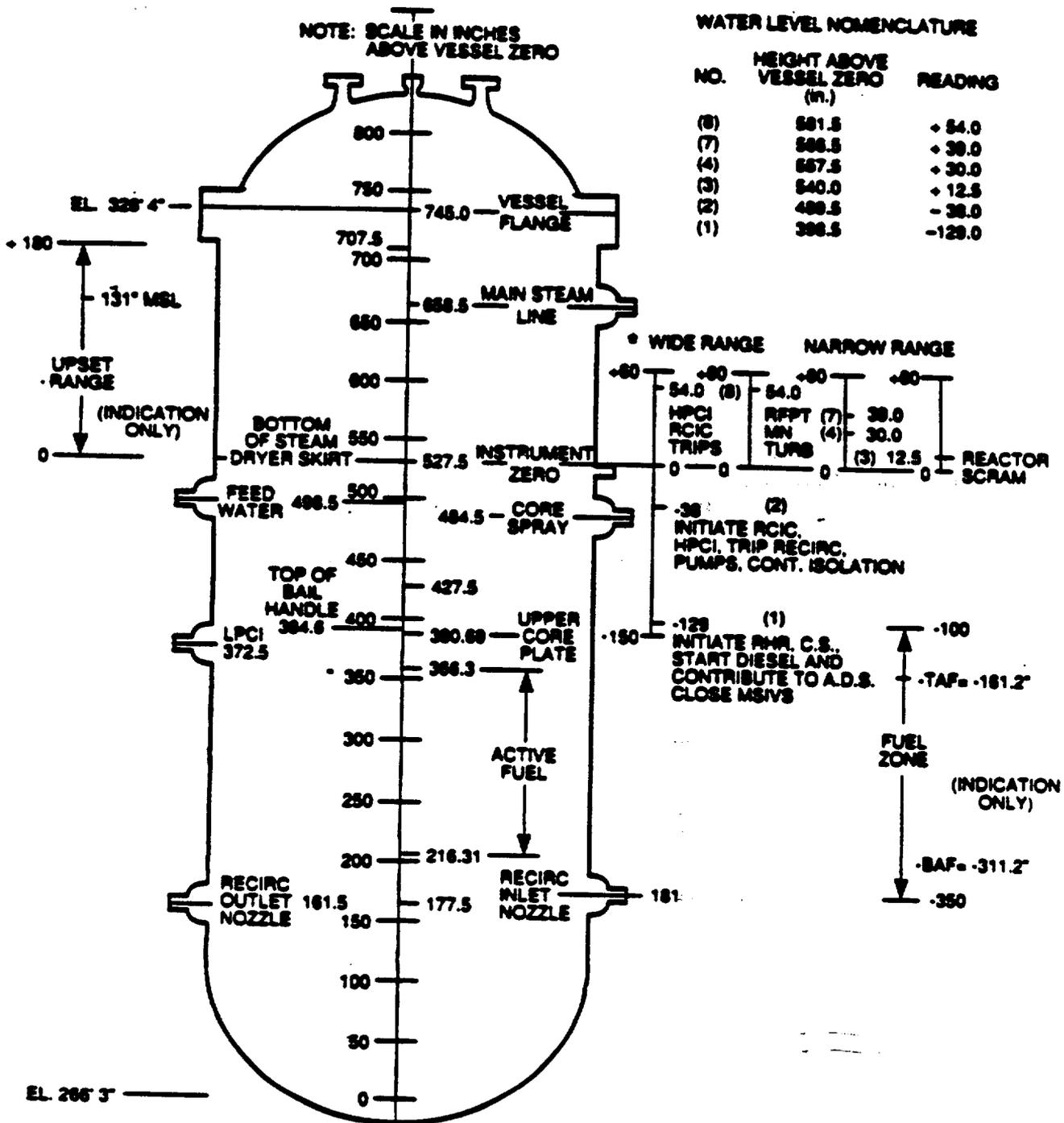
This instrumentation includes provisions for monitoring the concentrations of potentially explosive gas mixtures and noble gases in the off-gas system.

#### 3/4.3.8. TURBINE OVERSPEED PROTECTION SYSTEM

This specification is provided to ensure that the turbine overspeed protection system instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

#### 3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

The feedwater/main turbine trip system actuation instrumentation is provided to initiate action of the feedwater system/main turbine trip system in the event of failure of feedwater controller under maximum demand.



**\* Wide Range Level**

This indication is reactor coolant temperature sensitive. The calibration is thus made at rated conditions. The level error at low pressures (temperatures) is bounded by the safety analysis which reflects the weight-of-coolant above the lower tap, and not indicated level.

BASES FIGURE B 3/4.3-1  
REACTOR VESSEL WATER LEVEL



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION  
RELATED TO AMENDMENT NOS. 70 AND 33 TO FACILITY OPERATING  
LICENSE NOS. NPF-39 AND NPF-85  
PHILADELPHIA ELECTRIC COMPANY  
LIMERICK GENERATING STATION, UNITS 1 AND 2  
DOCKET NOS. 50-352 AND 50-353

1.0 INTRODUCTION

By letter dated May 6, 1993, as supplemented by letter dated April 18, 1994, the Philadelphia Electric Company (PECo or the licensee) submitted a request for changes to the Limerick Generating Station (LGS), Units 1 and 2, Technical Specifications (TS). The requested changes would extend surveillance test intervals (STIs) and allowed outage times (AOTs) for selected actuation instrumentation based on "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," GENE-770-06-1, February 1991, and as approved by NRC in "Review of BWR Owners Group Report GENE-770-06-1 on Justification for Extending Surveillance Test Intervals and Allowed Out-of-Service Times for Selected BWR Actuation Instrumentation" dated July 21, 1992. In addition, PECo proposes several editorial changes that correct previous omissions and reformat the TS. The supplemental letter does not change the proposed no significant hazards determination.

2.0 BACKGROUND

Licensing Topical Report (LTR), "BWR Owners Group Response to NRC Generic Letter 83-28, Item 4.5.3," General Electric Company, NEDC-30844, January 1985, provided justification for the acceptability of current Reactor Protection System (RPS) instrumentation. In addition, the same report established a basis for extending STIs and AOTs for RPS instrumentation based on reliability analyses which estimate RPS instrumentation failure frequency. The analyses were further developed in GENE-770-06-1 for extending TS STIs and AOTs for selected Boiling Water Reactor (BWR) Actuation Instrumentation, and the results were subsequently approved as detailed in an NRC safety evaluation dated July 21, 1992. The staff's evaluation described the acceptability of both the analyses and the proposed TS changes provided to the NRC. In addition, it provided criteria for plant-specific implementation of the generically-approved TS changes. The proposed editorial changes correct omissions to TS Sections 4.1.3.1.4 and 4.6.2.1, as a result of Change Request No. 89-16, dated April 26, 1990, and reformat TS 4.6.2.1(c) in two separate sections to identify currently proposed TS changes.

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### 3.0 EVALUATION

The licensee has stated that the effect on safety of the proposed extensions to the STIs and AOTs of the selected BWR actuation instrumentation has been addressed in GENE-770-06-1. Furthermore, the NRC has detailed their acceptance of the analyses and the conclusions of GENE-770-06-1 in the July 21, 1992, safety evaluation. The staff's evaluation concludes that implementation of the TS changes proposed in GENE-770-06-1 would provide an overall enhancement to plant safety and that the proposed changes are acceptable subject to the licensee documenting (1) plant-specific applicability, and (2) that instrument drift is bounded by the assumptions of the generic analyses. The licensee has addressed the acceptance conditions as follows:

1. The licensee has conducted a plant-specific review of the applicability of the GENE-770-06-1. For the selected BWR actuation instrumentation, the review compared the LGS configuration with those in the GENE-770-06-1 analyses. This comparison concluded that the configurations are consistent with those in the GENE-770-06-1 analyses and thus the analyses are applicable to LGS, Units 1 and 2.
2. In 1988, the NRC issued additional guidance regarding instrument drift. The staff guidance letter stated that "licensees need only confirm that the setpoint drift, which could be expected under the extended STIs has been studied and either (1) has been shown to remain within the existing allowance in the RPS (for BWRs)... instrument setpoint calculation, or (2) that the allowance and setpoint have been adjusted to account for the additional expected drift." Present setpoint calculations for LGS are based on an 18-month calibration interval. Therefore, drift occurring during a 3-month STI falls within the existing drift allowance. To further verify this, the licensee has examined the instrument drift data of over three consecutive monthly test intervals. "Limerick Generating Station, Unit 1, Instrument Drift Data for Selected Boiling Water Reactor Actuation Instrumentation," provides the as-found drift data on a 15% sample of affected LGS, Unit 1 instrumentation. This data provides actual verification that the drift occurring over three consecutive test intervals (i.e., one calendar quarter) is within the acceptable limits. Further, the basis for this proposed change is not affected by the pending request to increase the STIs for certain instruments from a nominal 18-month frequency to a nominal 24-month frequency.

The staff finds that the licensee has demonstrated the applicability and acceptability of all proposed changes as stated above. Therefore, the staff concludes that the changes proposed will minimize testing and relax restrictive AOTs, while providing overall plant safety.

The proposed editorial changes that correct omissions for TS Change Request No. 89-16 and reformat TS 4.6.2.1(c) are found acceptable. In TS Change Request No. 89-16, the licensee failed to include the necessary mark-up of TS Sections 4.1.3.1.4 and 4.6.2.1. Therefore, the licensee is proposing to correct these omissions to eliminate the inconsistency between the 31-day STI currently required by TS Sections 4.1.3.1.4 and 4.6.2.1 and those which were approved in Amendments 53 and 17, currently shown in TS Tables 4.3.1.1-1 and 4.3.3.1-1, respectively. The editorial change to TS 4.6.2.1(c) involves splitting this item into two separate items since the proposed changes apply only to the suppression chamber water level indicator and not the temperature indicators.

#### 4.0 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.

#### 5.0 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 (58 FR 34087). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

#### 6.0 CONCLUSION

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

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