Docket File

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

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: EXTENDED SURVEILLANCE TEST INTERVALS AND ALLOWED OUTAGE TIMES FOR CONTAINMENT ISOLATION ACTUATION INSTRUMENTATION, LIMERICK GENERATING STATION, UNITS 1 AND 2 (TAC NOS. M86308 AND M86309)

The Commission has issued the enclosed Amendment No. 69 to Facility Operating License No. NPF-39 and Amendment No. 32 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 April 19, 1993, as supplemented by letter dated April 18, 1994.

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

CP-1

DF01

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

Enclosures:
1. Amendment No. 69 to
   License No. NPF-39
   Amendment No. 32 to
   License No. NPF-85
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION:

Docket File	MO'Brien(2)	ACRS(10)
NRC & Local PDRs	FRinaldi/JShea	OPA
PDI-2 Reading	OGC	OC/LFDCB
SVarga	DHagan, 3206	EWenzinger, RGN-I
JCalvo	GHill(4), P1-22	CAnderson, RGN-I
CMiller	CGrimes, 11E-21	JWermiel, 8H-3

*PREVIOUS CONCURRENCE

OFC	:PDI-2/LA	:PDI-2/PM	:*HICB	: OGC	:PDI-2/D	:	:
NAME	:MO'Brien	:FRinaldi:rb	:JWermiel	: EHoller	:CMiller	:	:
DATE	:4/25/93	:3/21/93	:08/31/93	:3/24/94	:4/26/94	:	:

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

Sincerely,

Frank Rinaldi

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

Enclosures:
1. Amendment No. 69 to
   License No. NPF-39
   Amendment No. 32 to
   License No. NPF-85
2. Safety Evaluation

cc w/enclosures:
See next page

cc:

J. W. Durham, Sr., Esquire
Sr. V.P. & General Counsel
PECO Energy Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Mr. Rod Krich, 52A-5
PECO Energy Company
955 Chesterbrook Boulevard
Wayne, Pennsylvania 19087-5691

Mr. David R. Helwig, Vice President
Limerick Generating Station
Post Office Box A
Sanatoga, Pennsylvania 19464

Mr. Robert Boyce
Plant Manager
Limerick Generating Station
P.O. Box A
Sanatoga, Pennsylvania 19464

Regional Administrator
U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Mr. Neil S. Perry
Senior Resident Inspector
US Nuclear Regulatory Commission
P. O. Box 596
Pottstown, Pennsylvania 19464

Mr. Craig L. Adams
Superintendent - Services
Limerick Generating Station
P.O. Box A
Sanatoga, Pennsylvania 19464

Mr. Rich R. Janati, Chief
Division of Nuclear Safety
PA Dept. of Environmental Resources
P. O. Box 8469
Harrisburg, Pennsylvania 17105-8469

Mr. James A. Muntz
Superintendent-Technical
Limerick Generating Station
P. O. Box A
Sanatoga, Pennsylvania 19464

Mr. James L. Kantner
Regulatory Engineer
Limerick Generating Station
P. O. Box A
Sanatoga, Pennsylvania 19464

Library
US Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Mr. Larry Hopkins
Superintendent-Operations
Limerick Generating Station
P. O. Box A
Sanatoga, Pennsylvania 19464

Mr. John Doering, Chairman
Nuclear Review Board
PECO Energy Company
955 Chesterbrook Boulevard
Mail Code 63C-5
Wayne, Pennsylvania 19087

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. 69
License No. NPF-39

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

   A. The application for amendment by Philadelphia Electric Company (the licensee) dated April 19, 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. 69 , 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

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. 69

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
xvii	xvii*
xviii	xviii
xix	xix
xx	xx*
3/4 3-9	3/4 3-9
3/4 3-10	3/4 3-10*
3/4 3-15	3/4 3-15*
3/4 3-16	3/4 3-16
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 3-32	3/4 3-32*
B 3/4 3-1	B 3/4 3-1*
B 3/4 3-2	B 3/4 3-2
B 3/4 3-3	B 3/4 3-3
B 3/4 3-4	B 3/4 3-4*

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

effective January 2, 1990

BASES

SECTION		PAGE

BASES

BASES

SECTION	PAGE

REACTOR COOLANT SYSTEM (Continued)

3/4.3.2. ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

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

b) With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirements for one trip system:

1. If placing the inoperable channel(s) in the tripped condition would cause an isolation, the inoperable channel(s) shall be restored to OPERABLE status within 6 hours. If this cannot be accomplished, the ACTION required by Table 3.3.2-1 for the affected trip function shall be taken, or the channel shall be placed in the tripped condition.

or

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

   a) 12 hours for trip functions common* to RPS Instrumentation.

   b) 24 hours for trip functions not common* to RPS Instrumentation.

The provisions of Specification 3.0.4 are not applicable.

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

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

effective December 17, 1991

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

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

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION
ACTION STATEMENTS

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

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

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

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

ACTION 24 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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

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

TABLE NOTATIONS

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

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

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

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

(b) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter. Trip functions common to RPS Actuation Instrumentation are shown in Table 4.3.2.1-1. In addition, for the HPCI system and RCIC system isolation, provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is OPERABLE and all required actuation instrumentation for that valve is OPERABLE, one channel may be placed in an inoperable status for up to 8 hours for required surveillance without placing the channel or trip system in the tripped condition.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRE
1. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Low, Level 2	S	Q	R	1, 2, 3
2) Low, Low, Low - Level 1	S	Q	R	1, 2, 3
b. Main Steam Line Radiation## - High	S	Q	R	1, 2, 3
c. Main Steam Line Pressure - Low	S	Q	R	1
d. Main Steam Line Flow - High	S	Q	R	1, 2, 3
e. Condenser Vacuum - Low	S	Q	R	1, 2**, 3**
f. Outboard MSIV Room Temperature - High	S	Q	R	1, 2, 3
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	S	Q	R	1, 2, 3
h. Manual Initiation	N.A.	R	N.A.	1. 2, 3
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level## Low - Level 3	S	Q	R	1, 2, 3
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	S	Q	R	1, 2, 3
c. Manual Initiation	N.A.	R	N.A.	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRE
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. RWCS Δ Flow - High	S	Q	R	1, 2, 3
b. RWCS Area Temperature - High	S	Q	R	1, 2, 3
c. RWCS Area Ventilation Δ Temperature - High	S	Q	R	1, 2, 3
d. SLCS Initiation	N.A.	R	N.A.	1, 2, 3
e. Reactor Vessel Water Level Low, Low, - Level 2	S	Q	R	1, 2, 3
f. Manual Initiation	N.A.	R	N.A.	1, 2, 3
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Δ Pressure - High	S	Q	R	1, 2, 3
b. HPCI Steam Supply Pressure, Low	S	Q	R	1, 2, 3
c. HPCI Turbine Exhaust Diaphragm Pressure - High	S	Q	R	1, 2, 3
d. HPCI Equipment Room Temperature - High	S	Q	R	1, 2, 3
e. HPCI Equipment Room Δ Temperature - High	S	Q	R	1, 2, 3
f. HPCI Pipe Routing Area Temperature - High	S	Q	R	1, 2, 3
g. Manual Initiation	N.A.	R	N.A.	1, 2, 3
h. HPCI Steam Line Δ Pressure Timer	N.A.	Q	R	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRE
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Δ Pressure - High	S	Q	R	1, 2, 3
b. RCIC Steam Supply Pressure - Low	S	Q	R	1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure - High	S	Q	R	1, 2, 3
d. RCIC Equipment Room Temperature - High	S	Q	R	1, 2, 3
e. RCIC Equipment Room Δ Temperature - High	S	Q	R	1, 2, 3
f. RCIC Pipe Routing Area Temperature - High	S	Q	R	1, 2, 3
g. Manual Initiation	N.A.	R	N.A.	1, 2, 3
h. RCIC Steam Line Δ Pressure Timer	N.A.	Q	R	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRE
6. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Low - Level 2	S	Q	R	1, 2, 3
2) Low, Low, Low - Level 1	S	Q	R	1, 2, 3
b. Drywell Pressure## - High	S	Q	R	1, 2, 3
c. North Stack Effluent Radiation - High	S	Q	R	1, 2, 3
d. Deleted				
e. Reactor Enclosure Ventilation Exhaust Duct - Radiation - High	S	Q	R	1, 2, 3
f. Outside Atmosphere to Reactor Enclosure Δ Pressure - Low	N.A.	M	Q	1, 2, 3
g. Deleted				
h. Drywell Pressure - High/ Reactor Pressure - Low	S	Q	R	1, 2, 3
i. Primary Containment Instrument Gas to Drywell Δ Pressure - Low	N.A.	M	Q	1, 2, 3
j. Manual Initiation	N.A.	R	N.A.	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRE
7. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level Low, Low - Level 2	S	Q	R	1, 2, 3
b. Drywell Pressure## - High	S	Q	R	1, 2, 3
c.1. Refueling Area Unit 1 Ventilation Exhaust Duct Radiation - High	S	Q	R	*#
2. Refueling Area Unit 2 Ventilation Exhaust Duct Radiation - High	S	Q	R	*#
d. Reactor Enclosure Ventilation Exhaust Duct Radiation - High	S	Q	R	1, 2, 3
e. Outside Atmosphere To Reactor Enclosure Δ Pressure - Low	N.A.	M	Q	1, 2, 3
f. Outside Atmosphere To Refueling Area Δ Pressure - Low	N.A.	M	Q	*
g. Reactor Enclosure Manual Initiation	N.A.	R	N.A.	1, 2, 3
h. Refueling Area Manual Initiation	N.A.	R	N.A.	*

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

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

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

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

3/4 3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

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

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

c. With either ADS trip system subsystem inoperable, restore the inoperable trip system to OPERABLE status within:

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

2. 72 hours.

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

SURVEILLANCE REQUIREMENTS

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

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

4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function shown in Table 3.3.3-3 shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ECCS trip system.

BASES

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be absorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

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

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279 for nuclear power plant protection systems. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P, "Technical Specification Improvement Analyses for BWR Reactor Protection System," as approved by the NRC and documented in the NRC Safety Evaluation Report (SER) (letter to T. A. Pickens from A. Thadani dated July 15, 1987. The bases for the trip settings of RPS are discussed in the bases for Specification 2.2.1.

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

effective December 17, 1991

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance.

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

Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high of low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D. C. operated valves, a 3 second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 10-second diesel startup and the 3 second load center loading delay. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 13-second delay. It follows that checking the valve speeds and the 13-second time for emergency power establishment will establish the response time for the isolation functions.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

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

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.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel. This instrumentation does not provide actuation of any of the emergency core cooling equipment.

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

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits and Section 3/4.3 Instrumentation. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

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

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.7 MONITORING INSTRUMENTATION
3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels, and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with 10 CFR Part 50, Appendix A, General Design Criteria 19, 41, 60, 61, 63, and 64.

effective December 17, 1991

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. 32
License No. NPF-85

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

   A. The application for amendment by Philadelphia Electric Company (the licensee) dated April 19, 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-85 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 32 , 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

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. 32

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
xvii	xvii*
xviii	xviii
xix	xix
xx	xx*
3/4 3-9	3/4 3-9
3/4 3-10	3/4 3-10*
3/4 3-15	3/4 3-15*
3/4 3-16	3/4 3-16
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 3-32	3/4 3-32*
B 3/4 3-1	B 3/4 3-1*
B 3/4 3-2	B 3/4 3-2
B 3/4 3-3	B 3/4 3-3
B 3/4 3-4	B 3/4 3-4*

effective January 2, 1991

BASES

BASES

SECTION | PAGE

REACTOR COOLANT SYSTEM (Continued)

3/4.3.2. ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

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

b) With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirements for one trip system:

1. If placing the inoperable channel(s) in the tripped condition would cause an isolation, the inoperable channel(s) shall be restored to OPERABLE status within 6 hours. If this cannot be accomplished, the ACTION required by Table 3.3.2-1 for the affected trip function shall be taken, or the channel shall be placed in the tripped condition.

or

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

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

   b) 24 hours for trip functions not common* to RPS Instrumentation.

The provisions of Specification 3.0.4 are not applicable.

---

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

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

effective December 17, 1997

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

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

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION
ACTION STATEMENTS

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

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

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

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

ACTION 24 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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

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

TABLE NOTATIONS

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

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

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

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

(b) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter. Trip functions common to RPS Actuation Instrumentation are shown in Table 4.3.2.1-1. In addition, for the HPCI system and RCIC system isolation, provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is OPERABLE and all required actuation instrumentation for that valve is OPERABLE, one channel may be placed in an inoperable status for up to 8 hours for required surveillance without placing the channel or trip system in the tripped condition.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRE
1. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Low, Level 2	S	Q	R	1, 2, 3
2) Low, Low, Low - Level 1	S	Q	R	1, 2, 3
b. Main Steam Line Radiation ## - High	S	Q	R	1, 2, 3
c. Main Steam Line Pressure - Low	S	Q	R	1
d. Main Steam Line Flow - High	S	Q	R	1, 2, 3
e. Condenser Vacuum - Low	S	Q	R	1, 2**, 3**
f. Outboard MSIV Room Temperature - High	S	Q	R	1, 2, 3
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	S	Q	R	1, 2, 3
h. Manual Initiation	N.A.	R	N.A.	1. 2, 3
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level ## Low - Level 3	S	Q	R	1, 2, 3
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	S	Q	R	1, 2, 3
c. Manual Initiation	N.A.	R	N.A.	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRE
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. RWCS Δ Flow - High	S	Q	R	1, 2, 3
b. RWCS Area Temperature - High	S	Q	R	1, 2, 3
c. RWCS Area Ventilation Δ Temperature - High	S	Q	R	1, 2, 3
d. SLCS Initiation	N.A.	R	N.A.	1, 2, 3
e. Reactor Vessel Water Level Low, Low, - Level 2	S	Q	R	1, 2, 3
f. Manual Initiation	N.A.	R	N.A.	1, 2, 3
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Δ Pressure - High	S	Q	R	1, 2, 3
b. HPCI Steam Supply Pressure, Low	S	Q	R	1, 2, 3
c. HPCI Turbine Exhaust Diaphragm Pressure - High	S	Q	R	1, 2, 3
d. HPCI Equipment Room Temperature - High	S	Q	R	1, 2, 3
e. HPCI Equipment Room Δ Temperature - High	S	Q	R	1, 2, 3
f. HPCI Pipe Routing Area Temperature - High	S	Q	R	1, 2, 3
g. Manual Initiation	N.A.	R	N.A.	1, 2, 3
h. HPCI Steam Line Δ Pressure Timer	N.A.	Q	R	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRE
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Δ Pressure - High	S	Q	R	1, 2, 3
b. RCIC Steam Supply Pressure - Low	S	Q	R	1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure - High	S	Q	R	1, 2, 3
d. RCIC Equipment Room Temperature - High	S	Q	R	1, 2, 3
e. RCIC Equipment Room Δ Temperature - High	S	Q	R	1, 2, 3
f. RCIC Pipe Routing Area Temperature - High	S	Q	R	1, 2, 3
g. Manual Initiation	N.A.	R	N.A.	1, 2, 3
h. RCIC Steam Line Δ Pressure Timer	N.A.	Q	R	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRE
6. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Low - Level 2	S	Q	R	1, 2, 3
2) Low, Low, Low - Level 1	S	Q	R	1, 2, 3
b. Drywell Pressure ## - High	S	Q	R	1, 2, 3
c. North Stack Effluent Radiation - High	S	Q	R	1, 2, 3
d. Deleted				
e. Reactor Enclosure Ventilation Exhaust Duct - Radiation - High	S	Q	R	1, 2, 3
f. Outside Atmosphere to Reactor Enclosure Δ Pressure - Low	N.A.	M	Q	1, 2, 3
g. Deleted				
h. Drywell Pressure - High/ Reactor Pressure - Low	S	Q	R	1, 2, 3
i. Primary Containment Instrument Gas to Drywell Δ Pressure - Low	N.A.	M	Q	1, 2, 3
j. Manual Initiation	N.A.	R	N.A.	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRE
7. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level Low, Low - Level 2	S	Q	R	1, 2, 3
b. Drywell Pressure ## - High	S	Q	R	1, 2, 3
c.1. Refueling Area Unit 1 Ventilation Exhaust Duct Radiation - High	S	Q	R	*#
2. Refueling Area Unit 2 Ventilation Exhaust Duct Radiation - High	S	Q	R	*#
d. Reactor Enclosure Ventilation Exhaust Duct Radiation - High	S	Q	R	1, 2, 3
e. Outside Atmosphere To Reactor Enclosure Δ Pressure - Low	N.A.	M	Q	1, 2, 3
f. Outside Atmosphere To Refueling Area Δ Pressure - Low	N.A.	M	Q	*
g. Reactor Enclosure Manual Initiation	N.A.	R	N.A.	1, 2, 3
h. Refueling Area Manual Initiation	N.A.	R	N.A.	*

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

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

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

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

INSTRUMENTATION

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

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

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

c. With either ADS trip system subsystem inoperable, restore the inoperable trip system to OPERABLE status within:

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

   2. 72 hours.

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

SURVEILLANCE REQUIREMENTS

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

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

4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function shown in Table 3.3.3-3 shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ECCS trip system.

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be absorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

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

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279 for nuclear power plant protection systems. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P, "Technical Specification Improvement Analyses for BWR Reactor Protection System," as approved by the NRC and documented in the NRC Safety Evaluation Report (SER) (letter to T. A. Pickens from A. Thadani dated July 15, 1987. The bases for the trip settings of RPS are discussed in the bases for Specification 2.2.1.

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

effective December 17, 1991

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance.

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

Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high of low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D. C. operated valves, a 3 second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 10-second diesel startup and the 3 second load center loading delay. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 13-second delay. It follows that checking the valve speeds and the 13-second time for emergency power establishment will establish the response time for the isolation functions.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

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

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.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel. This instrumentation does not provide actuation of any of the emergency core cooling equipment.

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

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits and Section 3/4.3 Instrumentation. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

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

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels, and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with 10 CFR Part 50, Appendix A, General Design Criteria 19, 41, 60, 61, 63, and 64.

effective December 17, 1991

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 69 AND 32 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 April 19, 1993, as supplemented by letter dated April 18, 1994, the Philadelphia Electric Company (the licensee) submitted a request for changes to the Limerick Generating Station, Units 1 and 2, Technical Specifications (TS). The requested changes would extend surveillance test intervals (STIs) and allowed outage times (AOTs) for containment isolation actuation instrumentation (IAI) as analyzed in, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," NEDC-31667P-A, July 1990, and as approved by NRC in the Safety Evaluation (SE), "Review of the BWR Owners Group Report NEDC-31667P on Justification for Extension of Surveillance Test Intervals and Allowed Outage Times for BWR Isolation Instrumentation Not Common to Reactor Protection System (RPS) or Emergency Core Cooling System (ECCS) Instrumentation," dated June 18, 1990. 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 the current RPS instrumentation STIs. In addition, it 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 NEDC-31677P-A, July 1990, for extending TS STIs and AOTs for the containment IAI, and the analyses were subsequently approved as detailed in the related NRC SE, dated June 18, 1990. This SE describes the acceptability of both the analyses and the proposed TS changes that were provided to the NRC in NEDC-31677P-A, July 1990. In addition, NRC's SE provided criteria for plant-specific implementation of the generically approved TS changes. Compliance with these plant-specific criteria is discussed in the evaluation.

3.0 EVALUATION

The proposed changes to extend STIs and AOTs for the containment IAI are consistent with those approved by the NRC and documented in the related SE, dated June 18, 1990. In addition, the identified administrative changes are required to implement the proposed AOT and STI changes.

The staff's SE of June 18, 1990, concluded that implementation of the TS changes proposed in NEDC-31667P-A, July 1990, would provide an overall enhancement to plant safety and that the changes were acceptable subject to requirement of the licensee's documentation of the following: (1) plant-specific applicability, and (2) that instrument drift is bounded by the assumptions of the generic analyses.

The licensee has conducted a plant-specific review of the applicability of the LTR to Limerick Generating Station, Units 1 and 2 (LGS). For the containment IAI, the review compared the LGS IAI configurations with those in the NEDC-31667P-A, July 1990, analyses. This comparison concluded that the configurations were consistent with those in the NEDC-31667P-A, July 1990, analyses and thus applicable to LGS.

The NRC has issued additional guidance regarding instrument drift in a letter dated April 27, 1988, "Staff Guidance for Licensee Determination that the Drift Characteristics for Instrumentation Used in RPS Channels are Bounded by NEDC-3085P Assumptions when the Functional Test Interval is Extended from Monthly to Quarterly." This letter states 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 instrument drift occurring during a 3-month STI falls within the existing drift allowance. Further, the licensee has stated that instrument drift data has been examined over three consecutive monthly test intervals to verify the above conclusion. Also, the licensee has enclosed the document, "Limerick Generating Station, Unit 2, Instrument Drift Data for Containment Isolation Actuation Instrumentation," which provides the as-found drift data on a ten percent (10%) sample of LGS Unit 2 IAI. This data provides actual verification that the drift occurring over three consecutive test intervals (i.e., one calendar quarter) is within acceptable limits.

In conclusion, the staff's SE of June 18, 1990, provided acceptable TS changes based on the referenced LTR. The licensee has proposed TS changes consistent with those previously approved and specifically designated by the staff, and those administrative changes necessary to properly implement the proposed changes. Therefore, the staff concludes that the NRC criteria for demonstrating the applicability and acceptability of all proposed changes has been met, as discussed above. The staff also concludes that the changes proposed will minimize unnecessary testing and relax excessively restrictive AOTs, which can provide an overall enhancement to plant safety. Further, the staff concludes that the administrative changes which are not addressed in the NRC SERs are necessary to correct inconsistencies and to facilitate implementation of the proposed TS changes.

The proposed administrative changes to TS Index pages "xviii" and "xix" are necessary to accurately reflect the location of various Sections in the TS. These changes are to correct inconsistencies and to reflect additions to the TS Bases which reference the appropriate LTR(s) and accompanying SER(s). Each of the TS Instrumentation Bases page changes are proposed to correct inconsistencies, make an addition, as just described, or to accommodate carry over from a previous page as a result of the addition. These changes have no impact on safety and, therefore, are acceptable to the staff.

The modifications to the notes referenced on TS pages 3/4 3-9, 3/4 3-16, and the associated notes in TS Table 4.3.2.1-1 are administrative as well and address the changes in STIs and AOTs addressed by this TS change. The notes are being revised to eliminate references which will no longer be necessary upon approval of the proposed changes. Specifically, the note is being revised to eliminate the previous references to instrumentation which was common to the Emergency Core Cooling Systems (ECCS) or the containment IAI, but which are no longer necessary. Modification of these notes does not affect any requirements of the TS, and thus has no impact on safety. Therefore, they are acceptable to the staff.

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 and change surveillance requirements. 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

34086). 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.

Principal Contributor: F. Rinaldi

Date: April 26, 1994