September 15, 1997

Ms. Irene Johnson, Acting Manager
Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M99125 AND M99126)

Dear Ms. Johnson:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 120 to Facility Operating License No. NPF-11 and Amendment No. 105 to Facility Operating License No. NPF-18 for the LaSalle County Station, Units 1 and 2, respectively. The amendments are in response to your application dated July 1, 1997.

The amendments revise Technical Specification definition 1.4, Channel Calibration, to allow an alternative method of calibrating thermocouples and resistance temperature detector sensors. The amendments also make editorial and administrative corrections to TS Table 3.3.2-1, Table 3.3.6-1, and Bases Section 3/4.3.1.

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

Sincerely,

Original Signed By:

Donna M. Skay, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-373, 50-374

Enclosures:
1. Amendment No. 120 to NPF-11
2. Amendment No. 105 to NPF-18
3. Safety Evaluation

cc w/encl: see next page

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Docket File	PUBLIC	PDIII-2 r/f	~~J. Roe, JWR~~
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Docket Nos. 50-373, 50-374

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OFFICE	PM:PDIII-2	e	LA:PDIII-2	e	D:PDIII-2	E	OGC		HICB	E
NAME	DSKAY		CMOORE		RCAPRA		RBachmann		JWERMEIL	
DATE	07/04/97		07/31/97		09/08/97		08/12/97		08/5/97	

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

September 15, 1997

Ms. Irene Johnson, Acting Manager
Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
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The amendments revise Technical Specification definition 1.4, Channel Calibration, to allow an alternative method of calibrating thermocouples and resistance temperature detector sensors. The amendments also make editorial and administrative corrections to TS Table 3.3.2-1, Table 3.3.6-1, and Bases Section 3/4.3.1.

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Sincerely,

Donna M. Skay

Donna M. Skay, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-373, 50-374

Enclosures:
1. Amendment No. 120 to NPF-11
2. Amendment No. 105 to NPF-18
3. Safety Evaluation

cc w/encl: see next page

cc:

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Sidley and Austin
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Chicago, Illinois 60603

Assistant Attorney General
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Suite 12
Chicago, Illinois 60601

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Marseilles, Illinois 61341-9756

Chairman
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LaSalle County Courthouse
Ottawa, Illinois 61350

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-373

LASALLE COUNTY STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 120
License No. NPF-11

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

   A. The application for amendment filed by the Commonwealth Edison Company (the licensee), dated July 1, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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 set forth in 10 CFR Chapter I;

   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 enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-11 is hereby amended to read as follows:

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(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 120 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Donna M. Skay

Donna M. Skay, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: September 15, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 120

FACILITY OPERATING LICENSE NO. NPF-11

DOCKET NO. 50-373

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain a vertical line indicating the area of change.

REMOVE	INSERT
1-1	1-1
3/4 3-13	3/4 3-13
3/4 3-51	3/4 3-51
B 3/4 3-1	B 3/4 3-1

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AVERAGE PLANAR EXPOSURE

1.2 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the required sensor, alarm, display, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping or total channel steps so that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is tested.

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION				
a. RHR Equipment Area Δ Temperature - High	8	1/RHR area	1, 2, 3	22
b. RHR Area Temperature - High	8	1/RHR area	1, 2, 3	22
c. RHR Heat Exchanger Steam Supply Flow - High	8	1	1, 2, 3	22
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	6	2	1, 2, 3	25
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	6	1	1, 2, 3	25
c. RHR Pump Suction Flow - High	6	1	1, 2, 3	25
d. RHR Area Temperature - High	6	1/RHR area	1, 2, 3	25
e. RHR Equipment Area ΔT - High	6	1/RHR area	1, 2, 3	25
B. MANUAL INITIATION				
1. Inboard Valves	1, 2, 5, 6, 7	1/group	1, 2, 3	26
2. Outboard Valves	1, 2, 5, 6	1/group	1, 2, 3	26
3. Inboard Valves	4(c)(e)	1/group	1, 2, 3 and **,#	26
4. Outboard Valves	4(c)(e)	1/group	1, 2, 3 and **,#	26
5. Inboard Valves	3, 8, 9	1/valve	1, 2, 3	26
6. Outboard Valves	3, 8, 9	1/valve	1, 2, 3	26
7. Outboard Valve	8(h)	1/group	1, 2, 3	26

TABLE 3.3.6-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. ROD BLOCK MONITOR(a)			
a. Upscale	2	1*	60
b. Inoperative	2	1*	60
c. Downscale	2	1*	60
2. APRM			
a. Flow Biased Simulated Thermal Power-Upscale	4	1	61
b. Inoperative	4	1, 2, 5	61
c. Downscale	4	1	61
d. Neutron Flux-High	4	2, 5	61
3. SOURCE RANGE MONITORS			
a. Detector not full in(b)	3	2	61
	2	5	61
b. Upscale(c)	3	2	61
	2	5	61
c. Inoperative(c)	3	2	61
	2	5	61
d. Downscale(d)	3	2	61
	2	5	61
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative	6	2, 5	61
d. Downscale(e)	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	2	1, 2, 5**	62
b. Scram Discharge Volume Switch in Bypass	1	5**	62
6. RECIRCULATION FLOW UNIT			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. Comparator	2	1	62

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 adsorbed 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, 1971, for nuclear power plant protection systems. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System", March 1988, and MDE-83-0485 Revision 3, "Technical Specification Improvement Analysis for the Reactor Protection System for LaSalle County Station, Units 1 and 2", April 1991. The bases for the trip settings of the PRS are discussed in the bases for Specification 2.2.1. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains RPS trip capability.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the accident analysis. The RPS RESPONSE TIME acceptance criteria are included in plant Surveillance procedures. Only those functions with times assumed in the accident analysis are required to be response time tested.

As stated in Note * of Table 3.3.1-2, Neutron detectors are exempt from response time testing. In addition, for Functional Units 3 and 4, per Note ##, the associated sensors are not required to be response time tested. For these

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-374

LASALLE COUNTY STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 105
License No. NPF-18

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

   A. The application for amendment filed by the Commonwealth Edison Company (the licensee), dated July 1, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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 set forth in 10 CFR Chapter I;

   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 enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-18 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 105 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Donna M. Skay

Donna M. Skay, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 15, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 105

FACILITY OPERATING LICENSE NO. NPF-18

DOCKET NO. 50-374

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain a vertical line indicating the area of change.

REMOVE	INSERT
1-1	1-1
3/4 3-13	3/4 3-13
3/4 3-51	3/4 3-51
B 3/4 3-1	B 3/4 3-1

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AVERAGE PLANAR EXPOSURE

1.2 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

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1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, display, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

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b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is tested.

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION				
a. RHR Equipment Area Δ Temperature - High	8	1/RHR area	1, 2, 3	22
b. RHR Area Temperature - High	8	1/RHR area	1, 2, 3	22
c. RHR Heat Exchanger Steam Supply Flow - High	8	1	1, 2, 3	22
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	6	2	1, 2, 3	25
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	6	1	1, 2, 3	25
c. RHR Pump Suction Flow - High	6	1	1, 2, 3	25
d. RHR Area Temperature - High	6	1/RHR area	1, 2, 3	25
e. RHR Equipment Area ΔT - High	6	1/RHR area	1, 2, 3	25
B. MANUAL INITIATION				
1. Inboard Valves	1, 2, 5, 6, 7	1/group	1, 2, 3	26
2. Outboard Valves	1, 2, 5, 6	1/group	1, 2, 3	26
3. Inboard Valves	4(c)(d)	1/group	1, 2, 3 and **,#	26
4. Outboard Valves	4(c)(e)	1/group	1, 2, 3 and **,#	26
5. Inboard Valves	3, 8, 9	1/valve	1, 2, 3	26
6. Outboard Valves	3, 8, 9	1/valve	1, 2, 3	26
7. Outboard Valve	8(h)	1/group	1, 2, 3	26

TABLE 3.3.6-1
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a. Flow Biased Simulated Thermal Power-Upscale	4	1	61
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d. Neutron Flux-High	4	2, 5	61
3. SOURCE RANGE MONITORS			
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b. Upscale(c)	3	2	61
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c. Inoperative(c)	3	2	61
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	2	5	61
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c. Inoperative	6	2, 5	61
d. Downscale(e)	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	2	1, 2, 5**	62
b. Scram Discharge Volume Switch in Bypass	1	5**	62
6. RECIRCULATION FLOW UNIT			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. Comparator	2	1	62

3/4.3 INSTRUMENTATION

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 adsorbed following a loss-of-coolant accident, and

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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, 1971, for nuclear power plant protection systems. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System", March 1988, and MDE-83-0485 Revision 3, "Technical Specification Improvement Analysis for the Reactor Protection System for LaSalle County Station, Units 1 and 2", April 1991. The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains RPS trip capability.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 120 TO FACILITY OPERATING LICENSE NO. NPF-11 AND

AMENDMENT NO. 105 TO FACILITY OPERATING LICENSE NO. NPF-18

COMMONWEALTH EDISON COMPANY

LASALLE COUNTY STATION, UNITS 1 AND 2

DOCKET NOS. 50-373 AND 50-374

1.0 INTRODUCTION

By letter dated July 1, 1997, Commonwealth Edison Company (ComEd, the licensee) submitted a request to modify the LaSalle, Units 1 and 2, Technical Specifications (TS). The proposed changes revise the definition of Channel Calibration and correct miscellaneous errors in the TS.

2.0 EVALUATION

2.1 Definition of Channel Calibration

The definition of Channel Calibration in Section 1.4 of the TS states that a calibration consists of the adjustment of the channel output such that it responds with the necessary range and accuracy. However, thermocouple and RTD sensors, which are required by TS to be calibrated, are not adjustable and in some cases not accessible. The current TS definition was derived from NUREG-0123, "BWR 5 Standard Technical Specifications." The licensee proposes to modify the definition to be consist with the wording in NUREG-1434, "Standard Technical Specifications, General Electric Plants, BWR/6."

The revised definition of Channel Calibration will include the following statement: "Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel." The licensee currently uses an inplace qualitative assessment for these sensors and, therefore, the change to the TS will not change current practice. The proposed change will allow compliance with the TS definition and is consistent with NUREG-1434. Therefore, the proposed change is acceptable.

2.2 Editorial Corrections

Manually Initiated Isolation Valves

TS Table 3.3.2-1, Isolation Actuation Instrumentation - B. Manual Initiation, lists the valve groups which are manually closed to provide primary

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containment isolation. TS 3.3.2 requires that a minimum number of channels of actuation instrumentation for these valves be operable. The table lists valve group 7 under both inboard and outboard valves. Table 4.3.2-1 requires a channel functional test for these valves on a refueling outage frequency. The licensee proposes to delete group 7 from the outboard manual isolation function.

Valve group 7 contains the valves for isolation of the Traversing Incore Probe (TIP) system. TIP system isolation consists of two valves outside of primary containment for each of five penetrations. The inboard isolation valve is a ball valve that automatically closes due to reactor vessel water level low and drywell pressure high. The inboard valves also close on a manual initiation signal and valve group 7 is correctly listed under B.1, Inboard Valves. The outboard isolation valves for the TIP system penetrations are explosive squib valves. These valves would be actuated using a keylock switch in the main control room if the TIP does not withdraw to allow closure of the associated ball valve. These valves are not actuated by the primary containment manual isolation logic since they are only actuated if the TIP fails to withdraw following receipt of a valid isolation actuation signal. Therefore, inclusion of these under TS 3/4.3.2 is inappropriate. The outboard isolation squib valves will continue to be required to be operable by TS 3.6.3, "Primary Containment Isolation Valves" and verified operable by TS 4.6.3.5. Inclusion of valve group 7 under B.2, Outboard Valves, is an administrative error that has existed since initial licensing. Based on the above, the deletion of valve group 7 from Table 3.3.2-1, Trip Function B.2, is acceptable.

IRM Rod Block

TS Table 3.3.6-1, "Control Rod Withdrawal Block Instrumentation", Trip Function 4.a, Intermediate Range Monitors (IRM), Detector not full-in, provides the applicable conditions and minimum operable channels for this rod block. The purpose of the IRM detector not-full-in rod block is to assure that no control rod is withdrawn during low neutron flux level operations (during refuel and startup modes) unless proper neutron monitoring capability is available in that all IRM detectors are correctly located. The Table currently has a note (e) associated with trip function 4.a. Note (e) states that this function shall be automatically bypassed when the IRM channels are on range 1. The licensee proposes to delete the reference to note (e) in trip function 4.a.

The IRM detector not-full-in rod block is required to be functioning whenever the IRMs are required to be operable, this is, in operational conditions 2 and 5. There is no automatic bypass installed for this rod block other than the reactor mode switch in Run position which bypasses all IRM trip functions. Therefore, note (e) has never been applicable to this trip function and its inclusion in this TS was an administrative error. Based on the above, the deletion of note (e) from Table 3.3.6-1, trip function 4.a, is acceptable.

Bases Change

TS Bases Section 3/4.3.1, Reactor Protection System Instrumentation, has a typographical error. The fifth paragraph of this section refers to Note #, which is applicable to Table 3.3.1-2, Functional Units 3 and 4. TS Table 3.3.1-2, Functional Units 3 and 4, are modified only by note ##. Therefore, the reference to Note # is a typographical error. Therefore, replacing Note # with Note ## in this Bases section is acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the 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 (62 FR 40848). 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.

5.0 CONCLUSION

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

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