March 16, 1995

Mr. D. L. Farrar
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. M90702 AND M90703)

Dear Mr. Farrar:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 102 to Facility Operating License No. NPF-11 and Amendment No. 87 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 October 24, 1994.

The amendments restructure the primary containment and primary containment leakage technical specifications to reduce the repetition of those requirements contained in NRC regulations such as Appendix J to 10 CFR Part 50. The amendments also support your requested exemptions from Appendix J requirements related to the scheduling of containment integrated leakage rate tests. The staff has addressed your requested exemptions separately from the issuance of these amendments. In addition to the restructuring and scheduling changes, the amendments incorporate the relocation of the list of primary containment isolation valves in accordance with Generic Letter 91-08 and a revision to the interval for functional testing of hydrogen recombiners in accordance with Generic Letter 93-05.

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
William D. Reckley

William D. Reckley, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-373 and 50-374

Enclosures:
1. Amendment No. 102 to NPF-11
2. Amendment No. 87 to NPF-18
3. Safety Evaluation

cc w/encl: see next page

DISTRIBUTION:

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DATE	01/24/95	01/20/95	01/17/95	[redacted]	02/28/95	03/16/95

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Note corrections

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D. L. Farrar
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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. 102
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 October 24, 1994, 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 paragraphs 2.C.(2) and 2.D* of Facility Operating License No. NPF-11 are hereby amended to read as follows:

*Page 16b is attached, for convenience, for the composite license to reflect this change.

2.C.(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 102 , 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.

2.D. The facility requires exemptions from certain requirements of 10 CFR Part 50, 10 CFR Part 70, and 10 CFR Part 73. These include:

(a) Exemptions from certain requirements of Appendices G, H and J and 10 CFR Part 73 are described in the Safety Evaluation Report and Supplement No. 1, No. 2 and No. 3 to the Safety Evaluation Report.

(b) An exemption was requested until the completion of the first refueling from the requirements of 10 CFR 70.24.

(c) An exemption from 10 CFR Part 50, Appendix E from performing a full scale exercise within one year before issuance of an operating license, both exemptions (b) and (c) are described in Supplement No. 2 of the Safety Evaluation Report.

(d) An exemption was requested from the requirements of 10 CFR 50.44 until either the required 100 percent rated thermal power trip startup test has been completed or the reactor has operated for 120 effective full power days as specified by the Technical Specifications. Exemption (d) is described in the safety evaluation of License Amendment No. 12.

(e) An exemption from the requirement of paragraph III.D of Appendix J to conduct the third Type A test of each ten-year service period when the plant is shutdown for the 10-year plant inservice inspections. Exemption (e) is described in the safety evaluation accompanying Amendment No.   to this license. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, and the rules and regulations of the Commission (except as hereinafter exempted therefrom), and the provisions of the Act.

3. This amendment is effective upon date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

William D Reckley

William D. Reckley, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachments:
1. License page 16b
2. Changes to the Technical Specifications

Date of Issuance: March 16, 1995

D. The facility requires exemptions from certain requirements of 10 CFR Part 50, 10 CFR Part 70, and 10 CFR Part 73. These include:

(a) Exemptions from certain requirements of Appendices G, H and J and 10 CFR Part 73 are described in the Safety Evaluation Report and Supplement No. 1, No. 2 and No. 3 to the Safety Evaluation Report.

(b) An exemption was requested until the completion of the first refueling from the requirements of 10 CFR 70.24.

(c) An exemption from 10 CFR Part 50, Appendix E from performing a full scale exercise within one year before issuance of an operating license, both exemptions (b) and (c) are described in Supplement No. 2 of the Safety Evaluation Report.

(d) An exemption was requested from the requirements of 10 CFR 50.44 until either the required 100 percent rated thermal power trip startup test has been completed or the reactor has operated for 120 effective full power days as specified by the Technical Specifications. Exemption (d) is described in the safety evaluation of License Amendment No. 12.

(e) An exemption from the requirement of paragraph III.D of Appendix J to conduct the third Type A test of each ten-year service period when the plant is shutdown for the 10-year plant inservice inspections. Exemption (e) is described in the safety evaluation accompanying Amendment No.   to this license. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, and the rules and regulations of the Commission (except as hereinafter exempted therefrom), and the provisions of the Act.

E. This license is subject to the following additional condition for the protection of the environment:

Before engaging in additional construction or operational activities which may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement and its Addendum, the licensee shall provide a written notification to the Director of the Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

F. Reporting to the Commission:

(a) The licensee shall report any violations of the requirements contained in Section 2, Items C(1), C(3) through (33), and E of this license within twenty-four (24) hours by telephone and confirmed by telegram, mailgram, or facsimile transmission to the NRC Regional Administrator, Region III, or designee, not later than the first working day following the violation, with a written followup report within fourteen (14) working days.

(b) The licensee shall notify the Commission, as soon as possible but not later than one hour, of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

G. The licensee shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

ATTACHMENT TO LICENSE AMENDMENT NO. 102

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. The page marked with an asterisk is provided for convenience.

REMOVE	INSERT
I	I
II	II
VII	VII
XXIII	XXIII
1-3	1-3
1-4	1-4
1-5	1-5
1-6	1-6
1-7	1-7
1-8	1-8
3/4 3-11	3/4 3-11
3/4 3-12	3/4 3-12
3/4 3-13	3/4 3-13
3/4 3-14	3/4 3-14
3/4 3-19	3/4 3-19
3/4 6-1	3/4 6-1
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-18	3/4 6-18
3/4 6-19	3/4 6-19
3/4 6-22	3/4 6-22
3/4 6-23	3/4 6-23
3/4 6-24	3/4 6-24
3/4 6-36	3/4 6-36
3/4 6-43	3/4 6-43
*3/4 6-44	*3/4 6-44
B 3/4 6-1	B 3/4 6-1
B 3/4 6-1a	B 3/4 6-1a
B 3/4 6-4	B 3/4 6-4
B 3/4 6-4a	B 3/4 6-4a

DEFINITIONS

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END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME (Continued)

breaker trip coil from when the monitored parameter exceeds its trip setpoint at the channel sensor of the associated:

a. Turbine stop valves, and
b. Turbine control valves.

The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

FRACTION OF LIMITING POWER DENSITY

1.14 The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.

FRACTION OF RATED THERMAL POWER

1.15 The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

FREQUENCY NOTATION

1.16 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

1.17 A GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.18 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

1.19 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

L_a

1.20 The maximum allowable primary containment leakage rate, L_a, shall be 0.635 % of primary containment air weight per day at the calculated peak containment pressure (P_a = 39.6 psig).

LIMITING CONTROL ROD PATTERN

1.21 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.22 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.23 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e, all relays and contacts, all trip units, solid state logic elements, etc. of a logic circuit, from sensor through and including the actuated device to verify OPERABILITY. THE LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.24 The MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be the highest value of the FLPD which exists in the core.

MEMBERS(S) OF THE PUBLIC

1.25 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

MINIMUM CRITICAL POWER RATIO

1.26 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

OFFSITE DOSE CALCULATION MANUAL

1.27 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Technical Specification Section 6.2.F.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-Annual Radioactive Effluent Release Reports required by Technical Specification Sections 6.6.A.3 and 6.6.A.4.

OPERABLE - OPERABILITY

1.28 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, a normal and an emergency electrical power source, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION - CONDITION

1.29 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

1.30 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.31 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.32 PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are open under administrative control as permitted by Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is OPERABLE pursuant to Specification 3.6.1.3.

d. The primary containment leakage rates are maintained within the limits per Surveillance Requirement 4.6.1.1.b.

e. The suppression chamber is OPERABLE pursuant to Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

g. Primary containment structural integrity has been verified in accordance with Surveillance Requirement 4.6.1.1.e.

PROCESS CONTROL PROGRAM

1.33 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.34 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.35 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3323 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.36 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE EVENT

1.37 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.38 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

1.39 SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All secondary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic damper secured in its closed position, except as provided in Table 3.6.5.2-1 of Specification 3.6.5.2.

b. All secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is OPERABLE pursuant to Specification 3.6.5.3.

d. At least one door in each access to the secondary containment is closed.

e. The sealing mechanism associated with each secondary containment penetration, e.g., welds, bellows or O-rings, is OPERABLE.

f. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.5.1.a.

SHUTDOWN MARGIN

1.40 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

SITE BOUNDARY

1.41 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SOURCE CHECK

1.42 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

STAGGERED TEST BASIS

1.43 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

1.44 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TURBINE BYPASS SYSTEM RESPONSE TIME

1.45 The TURBINE BYPASS SYSTEM RESPONSE TIME shall be time interval from when the turbine bypass control unit generates a turbine bypass valve flow signal until the turbine bypass valves travel to their required positions. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

UNIDENTIFIED LEAKAGE

1.46 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

VENTILATION EXHAUST TREATMENT SYSTEM

1.47 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.48 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
A. AUTOMATIC INITIATION				
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
(1) Low, Level 3	7	2	1, 2, 3	20
(2) Low Low, Level 2	2, 3	2	1, 2, 3	20
(3) Low Low Low, Level 1	1, 10	2	1, 2, 3	20
b. Drywell Pressure - High	2, 7, 10	2	1, 2, 3	20
c. Main Steam Line				
1) Radiation - High	1	2	1, 2, 3	21
	3	2	1, 2, 3	22
2) Pressure - Low	1	2	1	23
3) Flow - High	1	2/line(d)	1, 2, 3	21
d. Main Steam Line Tunnel Temperature - High	1	2	1(i)(j), 2(i)(j), 3(i)(j)	21
e. Main Steam Line Tunnel ΔTemperature - High	1	2	1(i)(j), 2(i)(j), 3(i)(j)	21
f. Condenser Vacuum - Low	1	2	1, 2*, 3*	21
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	4(c)(e)	2	1, 2, 3 and **	24
b. Drywell Pressure - High	4(c)(e)	2	1, 2, 3	24
c. Reactor Vessel Water Level - Low Low, Level 2	4(c)(e)	2	1, 2, 3, and #	24
d. Fuel Pool Vent Exhaust Radiation - High	4(c)(e)	2	1, 2, 3, and **	24

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
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	5	1	1, 2, 3	22
b. Heat Exchanger Area Temperature - High	5	1/heat exchanger	1, 2, 3	22
c. Heat Exchanger Area Ventilation ΔT - High	5	1/heat exchanger	1, 2, 3	22
d. SLCS Initiation	5(f)	NA	1, 2, 3	22
e. Reactor Vessel Water Level - Low Low, Level 2	5	2	1, 2, 3	22
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	8	1	1, 2, 3	22
b. RCIC Steam Supply Pressure - Low	8, 9(g)	2	1, 2, 3	22
c. RCIC Turbine Exhaust Diaphragm Pressure - High	8	2	1, 2, 3	22
d. RCIC Equipment Room Temperature - High	8	1	1, 2, 3	22
e. RCIC Steam Line Tunnel Temperature - High	8	1	1, 2, 3	22
f. RCIC Steam Line Tunnel Δ Temperature - High	8	1	1, 2, 3	22
g. Drywell Pressure - High	9(g)	2	1, 2, 3	22
h. RCIC Equipment Room Δ Temperature - High	8	1	1, 2, 3	22

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, 7	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 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION

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 - Close the affected system isolation valves within 1 hour and declare the affected system inoperable.

ACTION 23 - Be in at least STARTUP within 6 hours.

ACTION 24 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

ACTION 25 - Lock the affected system isolation valves closed within 1 hour and declare the affected system inoperable.

ACTION 26 - Provided that the manual initiation function is OPERABLE for each other group valve, inboard or outboard, as applicable, in each line, restore the manual initiation function to OPERABLE status within 24 hours; otherwise, restore the manual initiation function to OPERABLE status within 8 hours; otherwise:

a. Be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, or

b. Close the affected system isolation valves within the next hour and declare the affected system inoperable.

NOTES

* May be bypassed with reactor steam pressure ≤ 1043 psig and all turbine stop valves closed.

** When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

\# During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

(a) Deleted.

(b) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the channel in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter. In addition for those trip systems with a design providing only one channel per trip system, the channel may be placed in an inoperable status for up to 8 hours for required surveillance testing without placing the channel in the tripped condition provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is operable and all required actuation instrumentation for that redundant valve is OPERABLE, or place the trip system in the tripped condition.

(c) Also actuates the standby gas treatment system.

(d) A channel is OPERABLE if 2 of 4 instruments in that channel are OPERABLE.

(e) Also actuates secondary containment ventilation isolation dampers per Table 3.6.5.2-1.

(f) Closes only RWCU system inlet outboard valve.

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION	N/A
a. Reactor Vessel Water Level - Low, Level 3	
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	
c. RHR Pump Suction Flow - High	
d. RHR Area Cooler Temperature High	
e. RHR Equipment Area ΔT High	
B. MANUAL INITIATION	N/A
1. Inboard Valves	
2. Outboard Valves	
3. Inboard Valves	
4. Outboard Valves	
5. Inboard Valves	
6. Outboard Valves	
7. Outboard Valve	

TABLE NOTATIONS

* Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed.

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

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

N/A Not Applicable.

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all primary containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that are open under administrative control as permitted by Specification 3.6.3.

b. Perform required visual examinations and leakage rate testing except for primary containment air lock testing and main steam lines through the isolation valves, in accordance with and at the frequency# specified by 10 CFR 50, Appendix J. as modified by approved exemptions.

   The overall integrated leakage rate acceptance criterion is $\leq 1.0\ L_a$. The Type B and C combined leakage rate acceptance criterion is $\leq 0.60\ L_a$. However, during the first unit startup following testing performed in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the combined Type B and Type C tests, and $< 0.75\ L_a$ for the Type A test.

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*See Special Test Exception 3.10.1.

**Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been deinerted since the last verification or more often than once per 92 days.

#The provisions of Specification 4.02 are not applicable to the frequencies specified by 10 CFR 50, Appendix J.

SURVEILLANCE REQUIREMENTS (Continued)

c. By verifying each primary containment air lock OPERABLE per Specification 3.6.1.3.

d. By verifying the suppression chamber OPERABLE per Specification 3.6.2.1.

e. Verify primary containment structural integrity in accordance with the Inservice Inspection Program for Post Tensioning Tendons. The frequency shall be in accordance with the Inservice Inspection Program for Post Tensioning Tendons.

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 DELETED

Pages 3/4 6-3 and 3/4 6-4 Deleted

SURVEILLANCE REQUIREMENTS (Continued)

c. By verifying at least two suppression chamber water level instrumentation channels and at least 14 suppression pool water temperature instrumentation channels, 7 in each of two divisions, OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,

2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

3. CHANNEL CALIBRATION at least once per 18 months.

The suppression chamber water level and suppression pool temperature alarm setpoint shall be:

a) High water level ≤ +2 inches*

b) Low water level ≥ -3 inches*

c) High temperature ≤ 105°F

d. By conducting drywell-to-suppression chamber bypass leak tests at least once per 18 months at an initial differential pressure of 1.5 psi and verifying that the A/√k calculated from the measured leakage is within the specified limit.

If any 1.5 psi leak test results in a calculated A/√k >20% of the specified limit, then the test schedule for subsequent tests shall be reviewed by the Commission.

If two consecutive 1.5 psi leak tests result in a calculated A/√k greater than the specified limit, then:

1. A 1.5 psi leak test shall be performed at least once per 9 months until two consecutive 1.5 psi leak tests result in the calculated A/√k within the specified limits, and

2. A 5 psi leak test, performed with the second consecutive successful 1.5 psi leak test, results in a calculated A/√k within the specified limit, after which the above schedule of once per 18 months for only 1.5 psi leak tests may be resumed.

If any required 5 psi leak test results in a calculated A/√k greater than the specified limit, then the test schedule for subsequent tests shall be reviewed by the Commission.

If two consecutive 5 psi leak tests reult in a calculated A/√k greater than the specified limit, then a 5 psi leak test shall be performed at least once per 9 months until two consecutive 5 psi leak tests result in a calculated A/√k within the specified limit, after which the above schedule of once per 18 months for only 1.5 psi leak tests may be resumed.

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*Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

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3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 Each primary containment isolation valve and reactor instrumentation line excess flow check valve shall be OPERABLE**.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more of the primary containment isolation valves, except the reactor instrumentation line excess flow check valves, inoperable:

   1. Maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either;

      a) Restore the inoperable valve(s) to OPERABLE status, or

      b) Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position,* or

      c) Isolate each affected penetration by use of at least one closed manual valve or blind flange.*

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

b. With one or more of the reactor instrumentation line excess flow check valves inoperable:

   1. Operation may continue and the provisions of Specification 3.0.3 are not applicable provided that within 4 hours either:

      a) The inoperable valve is returned to OPERABLE status, or

      b) The instrument line is isolated and the associated instrument is declared inoperable.

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

*Isolation valves closed to satify these requirements may be reopened on an intermittent basis under administrative control.

**Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

4.6.3.2 Each primary containment automatic isolation valve shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each primary containment power operated or automatic isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 Each reactor instrumentation line excess flow check valve shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow.

4.6.3.5 Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing the explosive squib from at least one explosive valve such that the explosive squib in each explosive valve will be tested at least once per 90 months, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No explosive squib shall remain in use beyond the expiration of its shelf-life and operating-life.

4.6.3.6 At least once per 18 months:

a. Verify leakage rate through all four main steam lines through the isolation valves is $\leq$ 100 scfh when tested at $\geq$ 25.0 psig.**

b. Verify combined leakage rate of $\leq$ 1 gpm times the total number of primary containment isolation valves through hydrostatically tested lines that penetrate the primary containment is not exceeded when these isolation valves are tested at 1.1 P_a, $\geq$ 43.6 psig.*

* Results shall be excluded from the combined leakage for all penetrations and seals subject to Type B and C tests.

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SURVEILLANCE REQUIREMENTS (Continued)

4.6.4.2 The manual isolation valves on both sides of an inoperable and/or open suppression chamber-drywell vacuum breaker shall be verified to be closed at least once per 7 days.

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

DRYWELL AND SUPPRESSION CHAMBER HYDROGEN RECOMBINER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent drywell and suppression chamber hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one drywell and/or suppression chamber hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each drywell and suppression chamber hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 92 days by cycling each flow control valve and recirculation valve through at least one complete cycle of full travel.

b. At least once per 18 months by verifying, during a recombiner system functional test:

   1. That the heaters are OPERABLE by determining that the current in each phase differs by less than or equal to 5% from the other phases and is within 5% of the value observed in the original acceptance test, corrected for line voltage differences.

   2. That the reaction chamber gas temperature increases to 1200 $\pm$ 25°F within 2 hours.

c. At least once per 18 months by:

   1. Performing a CHANNEL CALIBRATION of all recombiner operating instrumentation and control circuits.

   2. Verifying the integrity of all heater electrical circuits by performing a resistance to ground test within 30 minutes following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 100,000 ohms.

DRYWELL AND SUPPRESSION CHAMBER OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.6.6.2 The drywell and suppression chamber atomosphere oxygen concentration shall be less than 4% by volume.

APPLICABILITY: OPERATIONAL CONDITION 1*, during the time period:

a. Within 24 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER, following startup, to

b. Within 24 hours prior to reducing THERMAL POWER to less than 15% of RATED THERMAL POWER, preliminary to a scheduled reactor shutdown.

ACTION:

With the oxygen concentration in the drywell and/or suppression chamber exceeding the limit, restore the oxygen concentration to within the limit within 24 hours or be in at least STARTUP within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.2 The oxygen concentration in the drywell and suppression chamber shall be verified to be within the limit within 24 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER and at least once per 7 days thereafter.

*See Special Test Exception 3.10.5.

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

The structural integrity of the primary containment is ensured by the successful completion of the Inservice Inspection Program for Post Tensioning Tendons and by associated visual inspections of the steel liner and penetrations for evidence of deterioration or breach of integrity. This ensures that the structural integrity of the primary containment will be maintained in accordance with the provisions of the Primary Containment Tendon Surveillance Program. Testing and Frequency are consistent with the recommendations of Regulatory Guide 1.35, Revision 3, except that the Unit 1 and 2 primary containments shall be treated as twin containments even though the Initial Structural Integrity Tests were not within 2 years of each other.

PRIMARY CONTAINMENT INTEGRITY is maintained by limiting overall integrated leakage to $\leq 1.0\ L_a$ and the Type B and C combined leakage rate acceptance criterion is $\leq 0.60\ L_a$. Prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test, the combined Type B and C leakage must be $< 0.60\ L_a$, and the overall Type A leakage must be $< 0.75\ L_a$ when a Type A test is scheduled. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

The maximum allowable leakage rate for the primary containment (L_a) is 0.635% by weight of the containment atmosphere per day at the calculated maximum peak containment pressure (P_a) of 39.6 psig.

Individual leakage rates specified for the primary containment air lock, main steam lines through the isolation valves, and valves in hydrostatically tested lines are addressed in LCO 3.6.1.3, and Surveillance Requirement 4.6.3.6.

Surveillance Requirement 4.6.1.1.b maintains PRIMARY CONTAINMENT INTEGRITY by requiring compliance with the visual examinations and leakage rate test requirements of 10 CFR 50, Appendix J, as modified by approved exemptions. Failure to meet air lock leakage testing (4.6.1.3) or main steam isolation valve leakage (4.6.3.6.a) does not necessarily result in a failure of this Surveillance Requirement, 4.6.1.1.b. The impact of the failure to meet these Surveillance Requirements 4.6.1.3 and 4.6.1.1.b must be evaluated against the Type A, B, and C acceptance criteria of 10 CFR 50, Appendix J, as modified by approved exemptions. The leakage limits for main steam lines through the isolation valves and leakage test results of Surveillance Requirement 4.6.3.6.a are not included in the total sum of Type B and C tests (approved exemption). As-left leakage prior to the first startup after

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY (Continued)

performing a required 10 CFR 50, Appendix J, leakage test is required to be $< 0.60\ L_a$ for combined Type B and C leakage, and $< 0.75\ L_a$ for overall Type A leakage. At all other times between required Type A tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. The combined Type B and C leakage remains as $\leq 0.60\ L_a$ between scheduled tests, in accordance with Appendix J.

The Frequency is required by 10 CFR 50, Appendix J, as modified by approved exemptions. Thus, 4.0.2 (which allows Frequency extensions) does not apply to Surveillance Requirement 4.6.1.1.b.

3/4.6.1.2 DELETED

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitation on closure and leak rate for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specification 3/4.6.1.1. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

BASES

DEPRESSURIZATION SYSTEMS (Continued)

Because of the large volume and thermal capability of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for the external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event of safety-relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety-relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety relief valve to assure mixing and uniformity of energy insertion to the pool

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

Primary Containment Isolation Valves (PCIVs) form a part of the primary containment boundary. The PCIV safety function is related to control primary containment leakage rates during accidents or other conditions to limit the untreated release of radioactive materials from the containment in excess of the design limits.

The automatic isolation valves are required to have isolation times within limits and actuate on an automatic isolation signal. The valves covered by this specification are listed with their associated stoke times, and other design information for lines penetrating the Primary Containment, in UFSAR Section 6.2.

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact.

Main steam lines through the isolation valves and hydrostatically tested valves must meet alternative leakage rate requirements. Other PCIV leakage rates are addressed by specification 3/4.6.1.1, "PRIMARY CONTAINMENT INTEGRITY". UFSAR Section 6.2 also describes special leakage test requirements and exemptions.

This specification provides assurance that the PCIVs will perform their designed safety functions to control leakage from the primary containment during accidents.

The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication

PRIMARY CONTAINMENT ISOLATION VALVES (Continued)

with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude assess to close the valves and that this action will prevent the release of radioactivity outside the primary containment.

Surveillance Requirement 4.6.3.6.a verifies leakage through all four main steam lines is $\leq$ 100 scfh when tested at $\geq P_t$ (25.0 psig). The transient and accident analyses are based on leakage at the specified leakage rate. The leakage rate for main steam lines through the isolation valves must be verified to be in accordance with the leakage test requirements of 10 CFR 50, Appendix J, as modified by approved exemptions. A Note has been added to this Surveillance Requirement requiring the results to be excluded from the total of Type B and Type C tests. This ensures that leakage rate for main steam lines through the isolation valves is properly accounted for in accordance with an approved exemption. The frequency is "at least once per 18 months" in accordance with an approved exemption.

Surveillance Requirement 4.6.3.6.b test of hydrostatically tested lines provides assurance that the assumptions of UFSAR Section 6.2 are met. The combined leakage rates must be demonstrated in accordance with the leakage rate test at a frequency of "at least once per 18 months". A Note has been added to this Surveillance Requirement requiring the results to be excluded the total of Type B and Type C tests. This is in accordance with 10 CFR 50, Appendix J, and approved exemptions.

3/4.6.4 VACUUM RELIEF

Vacuum relief breakers are provided to equalize the pressure between the suppression chamber and drywell. This system will maintain the structural integrity of the primary containment under conditions of large differential pressures.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There are four valves to provide redundancy so that operation may continue for up to 72 hours with one vacuum breaker inoperable provided that the manual isolation valves on each side are in the closed position.

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. 87
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 October 24, 1994, 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 paragraphs 2.C.(2) and 2.D* of Facility Operating License No. NPF-18 are hereby amended to read as follows:

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*Page 10 is attached, for convenience, for the composite license to reflect this change.

2.C.(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 87 , 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.

D. The facility requires exemptions from certain requirements of 10 CFR Part 50, 10 CFR Part 70, and 10 CFR Part 73. These include:

(a) Exemptions from certain requirements of Appendices G, H and J to 10 CFR Part 50, and to 10 CFR Part 73 are described in the Safety Evaluation Report and Supplement Numbers 1, 2, 3, and 5 to the Safety Evaluation Report.

(b) An exemption was requested until completion of the first refueling from the requirements of 10 CFR 70.24.

(c) An exemption from the requirement of paragraph III.D of Appendix J to conduct the third Type A test of each ten-year service period when the plant is shutdown for the 10-year plant inservice inspections.

(d) A one-time exemption from the requirement of paragraph III.A.6(b) of Appendix J to resume a Type A test schedule of three times in ten years. Exemptions (c) and (d) are described in the Safety Evaluation accompanying Amendment No. to this license. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted pursuant to 10 CFR 50.12. With the granting of these exemptions the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

3. This amendment is effective upon date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

William D Reckley

William D. Reckley, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachments:
1. License page 10
2. Changes to the Technical Specifications

Date of Issuance: March 16, 1995

D. The facility requires exemptions from certain requirements of 10 CFR Part 50, 10 CFR Part 70, and 10 CFR Part 73. These include:

(a) Exemptions from certain requirements of Appendices G, H and J to 10 CFR Part 50, and to 10 CFR Part 73 are described in the Safety Evaluation Report and Supplement Numbers 1, 2, 3, and 5 to the Safety Evaluation Report.

(b) An exemption was requested until completion of the first refueling from the requirements of 10 CFR 70.24.

(c) An exemption from the requirement of paragraph III.D of Appendix J to conduct the third Type A test of each ten-year service period when the plant is shutdown for the 10-year plant inservice inspections.

(d) A one-time exemption from the requirement of paragraph III.A.6(b) of Appendix J to resume a Type A test schedule of three times in ten years. Exemptions (c) and (d) are described in the Safety Evaluation accompanying Amendment No. to this license. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted pursuant to 10 CFR 50.12. With the granting of these exemptions the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. Before engaging in additional construction or operational activities which may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement and its Addendum, the licensee shall provide a written notification to the Director of the Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

F. With the exception of Section 2, Item C(2), the licensee shall report any violations of the requirements contained in Section 2.C and 2.E of this license within 24 hours by telephone and confirm by telegram, mailgram, or facsimile transmission to the NRC Regional Administrator, Region III, or that administrator's designee, no later than the first working day following the violation, with a written followup report within 14 days.

G. The licensee shall notify the Commission, as soon as possible but not later than one hour, of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

H. The licensee shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

ATTACHMENT TO LICENSE AMENDMENT NO. 87

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
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II	II
VII	VII
XXII	XXII
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1-4	1-4
1-5	1-5
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FRACTION OF LIMITING POWER DENSITY

1.14 The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.

FRACTION OF RATED THERMAL POWER

1.15 The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

FREQUENCY NOTATION

1.16 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

1.17 A GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.18 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

1.19 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

L_a

1.20 The maximum allowable primary containment leakage rate, L_a, shall be 0.635 % of primary containmenet air weight per day at the calculated peak containment pressure (Pa = 39.6 psig).

LIMITING CONTROL ROD PATTERN

1.21 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.22 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.23 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e., all relays and contacts, all trip units, solid state logic elements, etc. of a logic circuit, from sensor through and including the actuated device to verify OPERABILITY. THE LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.24 The MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be the highest value of the FLPD which exists in the core.

MEMBER(S) OF THE PUBLIC

1.25 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

MINIMUM CRITICAL POWER RATIO

1.26 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

OFFSITE DOSE CALCULATION MANUAL

1.27 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Technical Specification Section 6.2.F.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-Annual Radioactive Effluent Release Reports required by Technical Specification Sections 6.6.A.3 and 6.6.A.4.

OPERABLE - OPERABILITY

1.28 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, a normal and an emergency electrical power source, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION - CONDITION

1.29 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

1.30 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.31 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.32 PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are open under administrative control as permitted by Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is OPERABLE pursuant to Specification 3.6.1.3.

d. The primary containment leakage rates are maintained within the limits per Surveillance Requirement 4.6.1.1.b.

e. The suppression chamber is OPERABLE pursuant to Specification 3.6.2.1.

PRIMARY CONTAINMENT INTEGRITY (Continued)

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

g. Primary containment structural integrity has been verified in accordance with Surveillance Requirement 4.6.1.1.e.

PROCESS CONTROL PROGRAM

1.33 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.34 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.35 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3323 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.36 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE EVENT

1.37 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.38 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

1.39 SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All secondary containment penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, or deactivated automatic damper secured in its closed position, except as provided in Table 3.6.5.2-1 of Specification 3.6.5.2.

b. All secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is OPERABLE pursuant to Specification 3.6.5.3.

d. At least one door in each access to the secondary containment is closed.

e. The sealing mechanism associated with each secondary containment penetration, e.g., welds, bellows or O-rings, is OPERABLE.

f. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.5.1.a.

SHUTDOWN MARGIN

1.40 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

SITE BOUNDARY

1.41 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SOURCE CHECK

1.42 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

STAGGERED TEST BASIS

1.43 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.

STAGGERED TEST BASIS (Continued)

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

1.44 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TURBINE BYPASS SYSTEM RESPONSE TIME

1.45 The TURBINE BYPASS SYSTEM RESPONSE TIME shall be time interval from when the turbine bypass control unit generates a turbine bypass valve flow signal until the turbine bypass valves travel to their required positions. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

UNIDENTIFIED LEAKAGE

1.46 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

VENTILATION EXHAUST TREATMENT SYSTEM

1.47 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.48 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
A. AUTOMATIC INITIATION				
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
(1) Low, Level 3	7	2	1, 2, 3	20
(2) Low Low, Level 2	2, 3	2	1, 2, 3	20
(3) Low Low Low, Level 1	1, 10	2	1, 2, 3	20
b. Drywell Pressure - High	2, 7, 10	2	1, 2, 3	20
c. Main Steam Line				
1) Radiation - High	1	2	1, 2, 3	21
	3	2	1, 2, 3	22
2) Pressure - Low	1	2	1	23
3) Flow - High	1	2/line[(d)]	1, 2, 3	21
d. Main Steam Line Tunnel Temperature - High	1	2	1[(i)(j)], 2[(i)(j)], 3[(i)(j)]	21
e. Main Steam Line Tunnel ΔTemperature - High	1	2	1[(i)(j)], 2[(i)(j)], 3[(i)(j)]	21
f. Condenser Vacuum - Low	1	2	1, 2*, 3*	21
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	4[(c)(e)]	2	1, 2, 3 and **	24
b. Drywell Pressure - High	4[(c)(e)]	2	1, 2, 3	24
c. Reactor Vessel Water Level - Low Low, Level 2	4[(c)(e)]	2	1, 2, 3, and #	24
d. Fuel Pool Vent Exhaust Radiation - High	4[(c)(e)]	2	1, 2, 3, and **	24

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
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	5	1	1, 2, 3	22
b. Heat Exchanger Area Temperature - High	5	1/heat exchanger	1, 2, 3	22
c. Heat Exchanger Area Ventilation ΔT - High	5	1/heat exchanger	1, 2, 3	22
d. SLCS Initiation	5(f)	NA	1, 2, 3	22
e. Reactor Vessel Water Level - Low Low, Level 2	5	2	1, 2, 3	22
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	8	1	1, 2, 3	22
b. RCIC Steam Supply Pressure - Low	8, 9(g)	2	1, 2, 3	22
c. RCIC Turbine Exhaust Diaphragm Pressure - High	8	2	1, 2, 3	22
d. RCIC Equipment Room Temperature - High	8	1	1, 2, 3	22
e. RCIC Steam Line Tunnel Temperature - High	8	1	1, 2, 3	22
f. RCIC Steam Line Tunnel Δ Temperature - High	8	1	1, 2, 3	22
g. Drywell Pressure - High	9(g)	2	1, 2, 3	22
h. RCIC Equipment Room Δ Temperature - High	8	1	1, 2, 3	22

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, 7	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 3.3.2-1 (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 - Close the affected system isolation valves within 1 hour and declare the affected system inoperable.

ACTION 23 - Be in at least STARTUP within 6 hours.

ACTION 24 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

ACTION 25 - Lock the affected system isolation valves closed within 1 hour and declare the affected system inoperable.

ACTION 26 - Provided that the manual initiation function is OPERABLE for each other group valve, inboard or outboard, as applicable, in each line, restore the manual initiation function to OPERABLE status within 24 hours; otherwise, restore the manual initiation function to OPERABLE status within 8 hours; otherwise:

a. Be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, or
b. Close the affected system isolation valves within the next hour and declare the affected system in operable.

TABLE NOTATIONS

* May be bypassed with reactor steam pressure < 1043 psig and all turbine stop valves closed.

** When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

\# During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

(a) Deleted.

(b) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the channel in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter. In addition for those trip systems with a design providing only one channel per trip system, the channel may be placed in an inoperable status for up to 8 hours for required surveillance testing without placing the channel in the tripped condition provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is operable and all required actuation instrumentation for that redundant valve is OPERABLE, or place the trip system in the tripped condition.

(c) Also actuates the standby gas treatment system.

(d) A channel is OPERABLE if 2 of 4 instruments in that channel are OPERABLE.

(e) Also actuates secondary containment ventilation isolation dampers per Table 3.6.5.2-1.

(f) Closes only RWCU system inlet outboard valve.

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION	N/A
a. Reactor Vessel Water Level - Low, Level 3	
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	
c. RHR Pump Suction Flow - High	
d. RHR Area Cooler Temperature High	
e. RHR Equipment Area ΔT High	
B. MANUAL INITIATION	N/A
1. Inboard Valves	
2. Outboard Valves	
3. Inboard Valves	
4. Outboard Valves	
5. Inboard Valves	
6. Outboard Valves	
7. Outboard Valve	

TABLE NOTATIONS

* Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed.

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

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

N/A Not Applicable.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all primary containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that are open under administrative control as permitted by Specification 3.6.3.

b. Perform required visual examinations and leakage rate testing except for primary containment air lock testing and main steam lines through the isolation valves, in accordance with and at the frequency# specified by 10 CFR 50, Appendix J, as modified by approved exemptions.

   The overall integrated leakage rate acceptance criterion is $\leq$ 1.0 L_a. The Type B and C combined leakage rate acceptance criterion is $\leq$ 0.60 L_a. However, during the first unit startup following testing performed in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, the leakage rate acceptance criteria are < 0.60 L_a for the combined Type B and Type C tests, and < 0.75 L_a for the Type A test.

*See Special Test Exception 3.10.1

**Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been deinerted since the last verification or more often than once per 92 days.

#The provisions of Specification 4.0.2 are not applicable to the frequencies specified by 10 CFR 50, Appendix J.

SURVEILLANCE REQUIREMENTS (Continued)

c. By verifying each primary containment air lock OPERABLE per Specification 3.6.1.3.

d. By verifying the suppression chamber OPERABLE per Specification 3.6.2.1.

e. Verify primary containment structural integrity in accordance with the Inservice Inspection Program for Post Tensioning Tendons. The frequency shall be in accordance with the Inservice Inspection Program for Post Tensioning Tendons.

PRIMARY CONTAINMENT LEAKAGE

3.6.1.2 Deleted.

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SURVEILLANCE REQUIREMENTS (Continued)

c. By verifying at least two suppression chamber water level instrumentation channels and at least 14 suppression pool water temperature instrumentation channels, 7 in each of two divisions, OPERABLE by performance of a:

   1. CHANNEL CHECK at least once per 24 hours,

   2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

   3. CHANNEL CALIBRATION at least once per 18 months.

   The suppression chamber water level and suppression pool temperature alarm setpoint shall be:

   a) High water level $\leq$ +2 inches*

   b) Low water level $\geq$ -3 inches*

   c) High temperature $\leq$105°F

d. By conducting drywell-to-suppression chamber bypass leak tests at least once per 18 months at an initial differential pressure of 1.5 psi and verifying that the $A/\sqrt{k}$ calculated from the measured leakage is within the specified limit.

   If any 1.5 psi leak test results in a calculated $A/\sqrt{k}$ >20% of the specified limit, then the test sschedule for subsequent tests shall be reviewed by the Commission.

   If two consecutive 1.5 psi leak tests result in a calculated $A/\sqrt{k}$ greater than the specified limit, then:

   1. A 1.5 psi leak test shall be performed at least once per 9 months until two consecutive 1.5 psi leak tests result in the calculated $A/\sqrt{k}$ within the specified limits, and

   2. A 5 psi leak test, performed with the second consecutive successful 1.5 psi leak test, results in a calculated $A/\sqrt{k}$ within the specified limit after which the above schedule of once per 18 months for only 1.5 psi leak tests may be resumed.

If any required 5 psi leak test results in a calculated $A/\sqrt{k}$ greater than the specified limit, then the test schedule for subsequent tests shall be reviewed by the Commission.

If two consecutive 5 psi leak tests reult in a calculated $A/\sqrt{k}$ greater than the specified limit, then a 5 psi leak test shall be performed at least once per 9 months until two consecutive 5 psi leak tests result in a calculated $A/\sqrt{k}$ within the specified limit, after which the above schedule of once per 18 months for only 1.5 psi leak tests may be resumed.

*Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

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3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 Each primary containment isolation valve and reactor instrumentation line excess flow check valve shall be OPERABLE**.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more of the primary containment isolation valves, except the reactor instrumentation line excess flow check valves, inoperable:

1. Maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either;

a) Restore the inoperable valve(s) to OPERABLE status, or

b) Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position,* or

c) Isolate each affected penetration by use of at least one closed manual valve or blind flange.*

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

b. With one or more of the reactor instrumentation line excess flow check valves inoperable:

1. Operation may continue and the provisions of Specification 3.0.3 are not applicable provided that within 4 hours either:

a) The inoperable valve is returned to OPERABLE status, or

b) The instrument line is isolated and the associated instrument is declared inoperable.

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

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*Isolation valves closed to satify these requirements may be reopened on an intermittent basis under administrative control.

**Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

4.6.3.2 Each primary containment automatic isolation valve shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each primary containment power operated or automatic isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 Each reactor instrumentation line excess flow check valve shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow.

4.6.3.5 Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing the explosive squib from at least one explosive valve such that the explosive squib in each explosive valve will be tested at least once per 90 months, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No explosive squib shall remain in use beyond the expiration of its shelf-life and operating-life.

4.6.3.6 At least once per 18 months:

a. Verify leakage rate through all four main steam lines through the isolation valves is $\leq$ 100 scfh when tested at $\geq$ 25.0 psig.*

b. Verify combined leakage rate of $\leq$ 1 gpm times the total number of primary containment isolation valves through hydrostatically tested lines that penetrate the primary containment is not exceeded when these isolation valves are tested at 1.1 P_a, $\geq$ 43.6 psig.*

* Results shall be excluded from the combined leakage for all penetrations and seals subject to Type B and C tests.

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SURVEILLANCE REQUIREMENTS (Continued)

4.6.4.2 The manual isolation valves on both sides of an inoperable and/or open suppression chamber-drywell vacuum breaker shall be verified to be closed at least once per 7 days.

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 cfm ± 10%.

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

DRYWELL AND SUPPRESSION CHAMBER HYDROGEN RECOMBINER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent drywell and suppression chamber hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one drywell and/or suppression chamber hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each drywell and suppression chamber hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 92 days by cycling each flow control valve and recirculation valve through at least one complete cycle of full travel.

b. At least once per 18 months by verifying, during a recombiner system functional test:

   1. That the heaters are OPERABLE by determining that the current in each phase differs by less than or equal to 5% from the other phases and is within 5% of the value observed in the original acceptance test, corrected for line voltage differences.

   2. That the reaction chamber gas temperature increases to 1200 ± 25°F within 2 hours.

c. At least once per 18 months by:

   1. Performing a CHANNEL CALIBRATION of all recombiner operating instrumentation and control circuits.

   2. Verifying the integrity of all heater electrical circuits by performing a resistance to ground test within 30 minutes following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 100,000 ohms.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

The structural integrity of the primary containment is ensured by the successful completion of the Inservice Inspection Program for Post Tensioning Tendons and by associated visual inspections of the steel liner and penetrations for evidence of deterioration or breach of integrity. This ensures that the structural integrity of the primary containment will be maintained in accordance with the provisions of the Primary Containment Tendon Surveillance Program. Testing and Frequency are consistent with the recommendations of Regulatory Guide 1.35, Revision 3, except that the Unit 1 and 2 primary containments shall be treated as twin containments even though the Initial Structural Integrity Tests were not within 2 years of each other.

PRIMARY CONTAINMENT INTEGRITY is maintained by limiting overall integrated leakage to $\leq 1.0\ L_a$ and the Type B and C combined leakage rate acceptance criterion is $\leq 0.60\ L_a$. Prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test, the combined Type B and C leakage must be $< 0.60\ L_a$, and the overall Type A leakage must be $< 0.75\ L_a$ when a Type A test is scheduled. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

The maximum allowable leakage rate for the primary containment (L_a) is 0.635% by weight of the containment atmosphere per day at the calculated maximum peak containment pressure (P_a) of 39.6 psig.

Individual leakage rates specified for the primary containment air lock, main steam lines through the isolation valves, and valves in hydrostatically tested lines are addressed in LCO 3.6.1.3, and Surveillance Requirement 4.6.3.6.

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY (Continued)

Surveillance Requirement 4.6.1.1.b maintains PRIMARY CONTAINMENT INTEGRITY by requiring compliance with the visual examinations and leakage rate test requirements of 10 CFR 50, Appendix J, as modified by approved exemptions. Failure to meet air lock leakage testing (4.6.1.3) or main steam isolation valve leakage (4.6.3.6.a) does not necessarily result in a failure of this Surveillance Requirement, 4.6.1.1.b. The impact of the failure to meet these Surveillance Requirements 4.6.1.3 and 4.6.1.1.b must be evaluated against the Type A, B, and C acceptance criteria of 10 CFR 50, Appendix J, as modified by approved exemptions. The leakage limits for main steam lines through the isolation valves and leakage test results of Surveillance Requirement 4.6.3.6.a are not included in the total sum of Type B and C tests (approved exemption). As-left leakage prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test is required to be $< 0.60\ L_a$ for combined Type B and C leakage, and $< 0.75\ L_a$ for overall Type A leakage. At all other times between required Type A tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. The combined Type B and C leakage remains as $\leq 0.60\ L_a$ between scheduled tests, in accordance with Appendix J.

The Frequency is required by 10 CFR 50, Appendix J, as modified by approved exemptions. Thus, 4.0.2 (which allows Frequency extensions) does not apply to Surveillance Requirement 4.6.1.1.b.

3/4.6.1.2 DELETED

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitation on closure and leak rate for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specification 3/4.6.1.1. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR 100 guidelines provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has

3/4.6.1 PRIMARY CONTAINMENT

MSIV LEAKAGE CONTROL SYSTEM (Continued)

occasionally occurred in the leak tightness of the MSIV's such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the isolation valves when isolation of the primary system and containment is required.

3/4.6.1.5 DELETED

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitation on drywell and suppression chamber internal pressure ensure that the containment peak pressure of 39.6 psig does not exceed the design pressure of 45 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 5 psid. The limit of 2.0 psig for initial positive primary containment pressure will limit the total pressure to 39.6 psig which is less that the design pressure and is consistent with the accident analysis.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during LOCA conditions and is consistent with the accident analysis.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The drywell and suppression chamber purge supply and exhaust isolation valves are required to be closed during plant operation except as required for inerting, de-inerting and pressure control. These valves have been demonstrated capable of closing during a LOCA or steam line break accident from the full open position.

DEPRESSURIZATION SYSTEMS (Continued)

Because of the large volume and thermal capability of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for the external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event of safety-relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety-relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety relief valve to assure mixing and uniformity of energy insertion to the pool

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

Primary Containment Isolation Valves (PCIVs) form a part of the primary containment boundary. The PCIV safety function is related to control primary containment leakage rates during accidents or other conditions to limit the untreated release of radioactive materials from the containment in excess of the design limits.

The automatic isolation valves are required to have isolation times within limits and actuate on an automatic isolation signal. The valves covered by this specification are listed with their associated stoke times, and other design information for lines penetrating the Primary Containment, in UFSAR Section 6.2.

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact.

Main steam lines through the isolation valves and hydrostatically tested valves must meet alternative leakage rate requirements. Other PCIV leakage rates are addressed by specification 3/4.6.1.1, "PRIMARY CONTAINMENT INTEGRITY". UFSAR Section 6.2 also describes special leakage test requirements and exemptions.

PRIMARY CONTAINMENT ISOLATION VALVES (Continued)

This specification provides assurance that the PCIVs will perform their designed safety functions to control leakage from the primary containment during accidents.

The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude assess to close the valves and that this action will prevent the release of radioactivity outside the primary containment.

Surveillance Requirement 4.6.3.6.a verifies leakage through all four main steam lines is ≤ 100 scfh when tested at ≥ P_t (25.0 psig). The transient and accident analyses are based on leakage at the specified leakage rate. The leakage rate for main steam lines through the isolation valves must be verified to be in accordance with the leakage test requirements of 10 CFR 50, Appendix J, as modified by approved exemptions. A Note has been added to this Surveillance Requirement requiring the results to be excluded from the total of Type B and Type C tests. This ensures that leakage rate for main steam lines through the isolation valves is properly accounted for in accordance with an approved exemption. The frequency is "at least once per 18 months" in accordance with an approved exemption.

Surveillance Requirement 4.6.3.6.b test of hydrostatically tested lines provides assurance that the assumptions of UFSAR Section 6.2 are met. The combined leakage rates must be demonstrated in accordance with the leakage rate test at a frequency of "at least once per 18 months". A Note has been added to this Surveillance Requirement requiring the results to be excluded the total of Type B and Type C tests. This is in accordance with 10 CFR 50, Appendix J, and approved exemptions.

3/4.6.4 VACUUM RELIEF

Vacuum relief breakers are provided to equalize the pressure between the suppression chamber and drywell. This system will maintain the structural integrity of the primary containment under conditions of large differential pressures.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There are four valves to provide redundancy so that operation may continue for up to 72 hours with one vacuum breaker inoperable provided that the manual isolation valves on each side are in the closed position.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 102 TO FACILITY OPERATING LICENSE NO. NPF-11 AND

AMENDMENT NO. 87 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 October 24, 1994, the Commonwealth Edison Company (ComEd, the licensee) requested changes to the Technical Specifications (TS) for the LaSalle County Station, Units 1 and 2, and exemptions from certain requirements of Appendix J to 10 CFR Part 50. The proposed amendments restructure the primary containment and primary containment leakage TSs to reduce the repetition of those requirements contained in Appendix J to 10 CFR Part 50. The amendments also support the licensee's requested exemptions from Appendix J requirements related to the scheduling of containment integrated leakage rate tests. The staff has addressed the requested exemptions separately from the review of these amendments. In addition to the restructuring and scheduling changes, the amendments incorporate the relocation of the list of primary containment isolation valves in accordance with Generic Letter 91-08 and a revision to the interval for functional testing of hydrogen recombiners in accordance with Generic Letter 93-05.

2.0 BACKGROUND

Section III.D.1(a) of Appendix J to 10 CFR Part 50 requires the performance of three Type A tests, overall integrated leakage rate tests (ILRTs), at approximately equal intervals during each 10-year service period with the third test of each set being conducted when the plant is shutdown for the 10-year plant inservice inspections. Section III.A.6(b) of Appendix J to 10 CFR Part 50 specifies additional requirements if two consecutive periodic Type A tests fail to meet the applicable acceptance criteria. The additional requirements entail performing Type A tests at each plant shutdown for refueling or eighteen month interval, whichever occurs first, until two consecutive Type A tests meet the acceptance criteria, after which, the testing schedule of Section III.D may be resumed. LaSalle County Station, Unit 2 experienced Type A test failures for the "as-found" condition at the first, third and fourth refueling outages as a result of penalties from local leak rate test (LLRT) (Type B and C) failures. Pursuant to the requirements of Section III.A.6(b), a Type A test was performed during the fifth refueling outage for Unit 2 and the results satisfied the applicable acceptance criteria.

The licensee has requested exemptions from the requirements of Section III.A.6(b) of Appendix J in support of not performing a Type A test during the sixth refueling outage for Unit 2, scheduled for early 1995. Specifically, the requested exemption is from the requirements of Section III.A.6(b) related to performing two consecutive successful tests prior to resuming the normal testing interval. In addition, the licensee has requested an exemption from Section III.D.1(a) of Appendix J which requires a Type A test during the 10-year inservice inspections because the sixth refueling outage is the last refueling outage of the first 10-year plant inservice inspection period. The licensee proposes to resume the testing interval of Section III.D based upon the successful test during the fifth refueling outage and the creation of a corrective action plan for Type C test failures and decouple the Type A test schedule from the inservice inspection period. The results of this proposal would be that the next scheduled Type A test would be performed during the seventh refueling outage for Unit 2 (currently scheduled for late 1996) in accordance with a test interval of 40 ± 10 months. The staff is addressing the licensee's request for exemption from the above requirements of Appendix J as a separate action from this proposed TS amendment.

The proposed amendment can be divided into three distinct technical areas. The first involves the restructuring of the primary containment integrity specification, TS 3/4.6.1, and primary containment leakage specification, TS 3/4.6.2, into a revised limiting condition for operation (LCO) and related surveillance requirements (SR). The resultant LCO and surveillances proposed by the licensee are similar to the content of NUREG-1433, Revision 0, "Standard Technical Specifications, General Electric Plants, BWR/4." The second significant change proposed by the licensee is the relocation of Table 3.6.3-1, "Primary Containment Isolation Valves," from the TS to plant procedures and the Updated Final Safety Analysis Report (UFSAR) in accordance with Generic Letter 91-08, "Removal of Component Lists from Technical Specifications." The third change involves a revision of the interval for functional testing of hydrogen recombiners from 6 months to 18 months in accordance with Generic Letter 93-05, "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation."

The change related to the relocation of the list of containment isolation valves as well as several aspects of the restructuring of the containment integrity specification involve relocation of existing specifications to licensee controlled documents such as the UFSAR. The Commission's regulatory requirements related to the content of TSs are set forth in 10 CFR 50.36. However, the regulation does not specify the particular requirements to be included in a plant's TSs. The Commission has provided guidance for the contents of TSs in its "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" ("Final Policy Statement"), 58 Federal Register 39132 (July 22, 1993), in which the Commission indicated that compliance with the Final Policy Statement satisfies Section 182a of the Atomic Energy Act. In particular, the Commission indicated that certain items could be relocated from the TS to licensee-controlled documents, consistent with the standard enunciated in *Portland General Electric Co.* (Trojan Nuclear

Plant), ALAB-531, 9 NRC 263, 273 (1979). In that case, the Atomic Safety and Licensing Appeal Board indicated that "technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety."

Consistent with this approach, the Final Policy Statement identified four criteria to be used in determining whether a particular matter is required to be included in the TSs, as follows: (1) installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary; (2) a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (4) a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety. As a result, existing TS requirements which fall within or satisfy any of the criteria in the Final Policy Statement must be retained in the TS, while those TS requirements which do not fall within or satisfy these criteria may be relocated to licensee-controlled documents.

3.0 EVALUATION

Primary Containment Integrity and Leakage Specifications

The evaluation of the changes to the primary containment integrity and leakage specifications will be presented by providing a description and evaluation for the proposed disposition for each of the affected individual requirements contained in the existing specifications. The existing TS 3/4.6 contains details that are also found in Appendix J to 10 CFR Part 50. The regulations require licensee compliance and can not be revised by the licensee. The proposed changes include the statement "... in accordance with and at the frequency specified by 10 CFR 50, Appendix J, as modified by approved exemptions." Therefore, direct reference to Appendix J eliminates the need for repetitious and unnecessary details within the TS. This is consistent with previous staff positions expressed in safety evaluations and the standard TSs and is considered to be administrative in nature. Therefore, detailed discussions or rationale will not be provided for those items removed from the TS based on the existence of similar requirements in Appendix J to 10 CFR Part 50.

- LCO 3.6.1.1, Primary Containment Integrity, and the associated Applicability and Actions are not changed by the proposed revisions.

- SR 4.6.1.1.a involves the performance of Type B tests after the closing of subject penetrations with the resultant leakage to be added to other Type B and C penetration leakages with the acceptance criteria for the combined leakage rate to be less than or equal to 0.60 L_a. The licensee proposes to delete this SR and states that the requirement is captured by the revised surveillance 4.6.1.1.b which requires testing "in accordance with and at the frequency specified by 10 CFR 50, Appendix J, as modified by approved exemptions." The acceptance criterion for the Type B and C combined leakage is retained in the proposed TS at the 0.60 L_a value. The staff finds that the requirements of the existing SR 4.6.1.1.a that are being deleted are repetitious of the requirements of Appendix J Section III, Leakage Testing Requirements. The change is, therefore, acceptable.

- SR 4.6.1.1.b is renumbered and changed to reflect the relocation of Table 3.6.3-1, Primary Containment Isolation Valves. This change is addressed in a later section of this evaluation.

- SRs 4.5.1.1.c, d, and e are unchanged.

- LCO 3.6.1.2, Primary Containment Leakage, defines limiting leakage rates for: (a) overall containment leakage (L_a), (b) combined leakage for listed penetrations, (c) leakage through the main steam isolation valves, and (d) combined leakage through valves which are hydrostatically tested. The licensee proposes to restructure the TS such that the current LCO regarding overall containment leakage is addressed by the revised 4.6.1.1.b which addresses testing in accordance with Appendix J. The numerical value for L_a is included in a TS Definition for L_a. The combined leakage for penetrations, 0.60 L_a, is captured by the revised surveillance 4.6.1.1.b. The specific limits for main steam isolation valves and hydrostatically tested valves are relocated to 3/4.6.3, Primary Containment Isolation Valves, as a new SR 4.6.3.6. The staff finds that the current requirements in LCO 3.6.1.2 are adequately addressed by the relocation of requirements within the TS as proposed by the licensee or are repetitious of the requirements explicitly provided in Appendix J. This change is acceptable.

- The existing Actions associated with various components of LCO 3.6.1.2 are to restore conditions to within the LCO requirements. This option is implicit for all LCOs. Therefore, this provision is unnecessary and omitting these actions is purely editorial. Since the LCO's are adequately addressed, as discussed above, and since the existing Actions do not contribute any additional requirements, the staff finds their deletion acceptable.

- SR 4.6.1.2.a dictates the normal Type A test interval of 40 ± 10 months with the third test of each set being conducted during the shutdown for the 10-year plant inservice inspections. This requirement is generally repetitious of the requirements of Section III.D.1(a) which requires three Type A tests, at approximately equal intervals, during each 10-year service period with the third test of each set performed during the 10-year plant inservice inspections. The only detail that the existing TS provides beyond repeating the Appendix J requirements is that the approximately equal interval is defined as 40 ± 10 months. The licensee has stated that this interval will be placed in the UFSAR and controlled in accordance with 10 CFR 50.59, Changes, Tests and Experiments. This detail is not considered to be a significant refinement of the requirements given in Appendix J and is supported by the numerous licensing amendments issued to extend the Type A test interval beyond 50 months. The staff finds that the Type A testing interval is adequately controlled by the wording contained in Appendix J, the inclusion of the 40 ± 10 month interval in the UFSAR and control of any changes to that stated interval band in accordance with 10 CFR 50.59.

- SR 4.6.1.2.b provides requirements for additional testing in the event of one or consecutive failures of Type A tests. The wording contained in the TS is repetitious of the requirements of Sections III.A.6(a) and (b) of Appendix J. This requirement is captured by the revised TS 4.6.1.1.b which states that testing shall be performed in accordance with and at the frequency specified by Appendix J, as modified by approved exemptions. The staff finds that the revised TS 4.6.1.1.b, with references to Appendix J and approved exemptions, adequately defines the requirements for additional testing following Type A test failures.

- SR 4.6.1.2.c requires the verification of the accuracy of each Type A test by the performance of a supplemental test. The requirement and criteria are repetitious of Section III.A.3(b) of Appendix J. Therefore, removal of the TS requirement is considered to be administrative or editorial in nature. The revised SR 4.6.1.1.b requiring testing in accordance with Appendix J, as modified by approved exemptions and the specific requirements of Appendix J Section III.A.3(b), adequately control the accuracy provisions of the deleted TS.

- SR 4.6.1.2.d requires Type B and C testing at intervals of no greater than 24 months except for tests involving air locks, main steam isolation valves, valves pressurized with seal fluids, and hydrostatically tested lines. These exceptions are addressed by other TS requirements (items 4.6.1.2.e-h) and previous exemptions from Appendix J. This existing TS requirement is repetitious of Section III.D.2(a) of Appendix J and the proposed deletion is, therefore, considered to be administrative in nature.

- SR 4.6.1.2.e specifies that, "Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3." The exception for the testing of air locks is specified in proposed SR 4.6.1.1.b as an exception to Appendix J. The specific requirements for the primary containment air lock for each unit are currently, and will remain, as specification 3/4.6.1.3, Primary Containment Air Locks. Also, the Definition of Primary Containment Integrity refers to specification 3.6.1.3. Therefore, the air lock testing and operability requirements are retained in TSs and are not affected by this proposed amendment. The staff finds the proposed change acceptable.

- Current SR 4.6.1.2.f requires special frequency requirements for leak rate testing for main steam lines through the isolation valves. The existing LCO 3.6.1.2.c, provides special acceptance criteria of "less than or equal to 100 scf per hour for all four main steam lines through the isolation valves when tested at 25.0 psig." The LCO is modified by footnote *, that states this is an "exemption to Appendix J of 10 CFR Part 50." This testing requirement, frequency, and acceptance criteria are relocated to specification 3/4.6.3, Primary Containment Isolation Valves, as proposed SR 4.6.3.6.a. This assures that the exemption concerning the main steam lines through the isolation valves is clearly separated from Type B and C combined leakage rate. The revised SR 4.6.1.1.b also reflects the current exemption of air locks and the main steam isolation valves from testing in accordance with Appendix J. The main steam line leakage through the isolation valves is excluded from the Type B and C combined leakage rate, because the main steam line leakage through the isolation valves is separately accounted for in the loss-of-coolant accident (LOCA) analysis for the dose consequences. The staff's approval of this exemption is documented in NUREG-0519, "Safety Evaluation Report Related to the Operation of LaSalle County Station, Units 1 and 2." To maintain this exemption, proposed SR 4.6.3.6.a is modified by a footnote. Therefore, the approved exemption regarding the leakage test requirements for main steam lines through the isolation valves is maintained as worded in proposed SR 4.6.3.6.a. The staff finds that the revised SR 4.6.3.6 properly maintains the requirements of the existing TS LCO and SR, and that there are no resultant changes in requirements. The proposed change is acceptable.

- SR 4.6.1.2.g concerning special testing requirements to be met for leakage from isolation valves that are sealed with fluid from a seal system, is consistent with Appendix J section III.C.3. Therefore, since proposed SR 4.6.1.1.b requires testing and test frequency in accordance with 10 CFR 50, Appendix J, and approved exemptions, the requirements are maintained and the changes are considered administrative or editorial in nature. The staff finds the changes acceptable.

- SR 4.6.1.2.h requires special frequency requirements for leak rate testing for ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment. The existing LCO 3.6.1.2.d, provides special acceptance criteria as follows: "A combined leakage rate of less than or equal to 1 gpm times the total number of ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 P_a, 43.6 psig." This testing requirement, frequency, and acceptance criteria are relocated to specification 3/4.6.3, Primary Containment Isolation Valves as proposed SR 4.6.3.6.b. This assures that the special surveillance frequency and acceptance criteria concerning ECCS and RCIC containment isolation valves in hydrostatically tested lines remain clearly separated from Type B and C combined leakage rate. The staff's approval of the exclusion of these valves when determining the combined leakage rate for all penetrations and valves as specified in Paragraph III.C.3 of Appendix J was documented in NUREG-0519. To maintain this exemption, proposed SR 4.6.3.6.b is modified by a footnote. The relocation of the requirements has no net effect on the testing or acceptable leakage through the affected valves. The staff finds the change acceptable.

- SR 4.6.1.2.i provides a statement that the provisions of TS 4.0.2 (maximum allowable extension of 25 percent for specified surveillance intervals) do not apply to the 24 month intervals associated with Type B and C testing or the 40 ± 10 month interval for Type A testing. The licensee proposes to maintain a similar statement, "The provisions of Specification 4.0.2 are not applicable to the frequencies specified by 10 CFR 50, Appendix J," as a footnote to revised SR 4.6.1.1.b. The staff finds it acceptable to include this footnote as a means to avoid confusion regarding the applicability of the 25 percent extension to the Appendix J defined intervals.

- SR 4.6.2.1.d governs the conduct of drywell-to-suppression chamber bypass leakage tests. The licensee's proposed changes reflect the deletion of those requirements which the TSs state may be discontinued in the event certain criteria are satisfied. Given the criteria for discontinuing the testing of existing SR 3.6.2.d.2 were satisfied, the licensee's proposed changes are editorial in that no changes in current testing requirements or acceptance criteria have been requested. The result of the changes is enhanced clarity by the deletion of outdated requirements that were only applicable during the early years of plant operation. The staff finds the changes acceptable.

- SR 4.6.4.3 requires that "vacuum breaker header flanges which have been broken shall be leak tested after remaking by Type B test at 39.6 psig per Specification 4.6.1.2.d." The licensee proposes to delete this TS SR because the flanges are within the primary containment penetration boundary for vacuum breaker penetrations and, therefore, subject to the requirements of Appendix J Section

III.D.2 which states "... If opened following a Type A or B test, containment penetrations subject to Type B testing shall be Type B tested prior to returning the reactor to an operating mode requiring containment integrity." The revised SR 4.6.1.1.b requiring testing in accordance with Appendix J as modified by approved exemptions and the specific requirements of Appendix J Section III.D adequately control the performance of Type B testing following restoration of the vacuum breaker header flanges. The staff finds the change acceptable.

- SR 4.6.6.1.d requires the measurement of the leakage rate associated with the drywell and suppression chamber hydrogen recombiner systems. The TS leakage testing is allowed to be performed as part of a Type A ILRT test or by measuring the leakage rate of the system outside the containment isolation valves. The drywell and suppression chamber hydrogen recombiner systems are designed to be part of the primary containment during an accident. Therefore, Appendix J requires the system to either be open to the primary containment during a Type A test, or accounted for by a separate leak rate test that is included in the overall integrated leakage test. The staff finds that the requirements of the existing SR 4.6.6.1.d that are being deleted, are repetitious of the requirements of Appendix J. The deletion of the SR from the TS is acceptable.

Relocation of Primary Containment Isolation Valve Table

In accordance with the guidance provided by Generic Letter 91-08, "Removal of Component Lists from Technical Specifications," the licensee has proposed to relocate Table 3.6.3-1, "Primary Containment Isolation Valves," from the TSs to Administrative Procedure and the UFSAR. Various editorial changes are also necessitated by the removal of the table (the disposition of table footnotes and various references to the table throughout the TSs). A review of the licensee's disposition of the existing footnotes to Table 3.6.3-1 found that the comments were either adequately addressed by other specifications, retained in Section 3.6.3, Primary Containment Isolation Valves, or could be deleted without affecting the existing requirements. Likewise, the editorial changes to address the reference to the table in various specifications were found to not to have any effect on the actual requirements contained in those specifications.

The staff's review of the proposed change determined that the relocation of Table 3.6.3-1 does not eliminate the requirements for the licensee to ensure containment isolation valves are capable of performing their safety function. Although Table 3.6.3-1 is relocated from the TSs to an Administrative Procedure and the UFSAR, the licensee must continue to evaluate changes in accordance with 10 CFR 50.59. Should the licensee's determination conclude that an unreviewed safety question is involved, due to either (1) an increase in the probability or consequences of accidents or malfunctions of equipment important to safety, (2) the creation of a possibility for an accident or malfunction of a different type than any evaluated previously, or (3) a reduction in the margin of safety, NRC approval and a license amendment would be required prior to implementation of the change. NRC inspection and enforcement programs also enable the staff to monitor facility changes and licensee adherence to UFSAR commitments and to take any remedial action that may be appropriate.

The staff's review concluded that 10 CFR 50.36 does not require the list of primary containment isolation valves to be retained in TSs. Requirements related to the operability, applicability, and surveillance requirements, including performance of testing to ensure operability of the isolation valves, are retained. However, the staff determined that the inclusion of list of isolation valves is an operational detail related to the licensee's safety analyses that are adequately controlled by the requirements of 10 CFR 50.59. Therefore, the continued processing of license amendments related to revisions of the table, where the revisions to those requirements do not involve an unreviewed safety question under 10 CFR 50.59, would afford no significant benefit with regard to protecting the public health and safety.

The staff has concluded, therefore, that relocation of Table 3.6.3-1 is acceptable because: (1) the inclusion of the table in TSs is not specifically required by 10 CFR 50.36 or other regulations; (2) the list of isolation valves has been relocated to an Administrative Procedure and the UFSAR, changes to the valves are adequately controlled by 10 CFR 50.59, and their inclusion in the TS is not required to avert an immediate threat to the public health and safety; and (3) changes that are deemed to involve an unreviewed safety question, will require prior NRC approval in accordance with 10 CFR 50.59(c).

Extension of Hydrogen Recombiner Functional Test Interval

The NRC has completed a comprehensive examination of SRs in the TSs that require testing during power operation. The evaluation is documented in NUREG-1366, "Improvements to Technical Specification Surveillance Requirements," dated December 1992. The staff found that while the majority of testing at power is important, safety can be improved, equipment degradation decreased, and an unnecessary burden on personnel resources eliminated by relaxing a small fraction of the TS testing intervals. Based on the results of the evaluations documented in NUREG-1366, the NRC issued Generic Letter 93-05, "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation," dated September 27, 1993.

Section 8.5 of NUREG-1366, discusses the recommended change in the functional test interval for hydrogen recombiners from once per six months to once per refueling interval. The staff based this recommendation on system redundancies included in plant designs for post-accident hydrogen control and the apparent high reliability of hydrogen recombiner systems.

The licensee proposed change to SR 4.6.6.1.b to change to the interval for performance of the hydrogen recombiner system functional test from six (6) months to eighteen (18) months is consistent with the recommendation in NUREG-1366. In addition to the studies referenced in NUREG-1366, the licensee stated that the plant specific history for the recombiners at LaSalle are consistent with the generic reliabilities found by the staff. Based upon the findings contained in NUREG-1366 and the plant specific evaluation performed by the licensee, the staff finds the proposed change to the hydrogen recombiner functional test interval from six months to eighteen months to be acceptable.

4.0 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.

5.0 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 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 (60 FR 498). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

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