April 5, 1996

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. M93597 AND M93598)

Dear Mr. Farrar:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 112 to Facility Operating License No. NPF-11 and Amendment No. 97 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 August 28, 1995, as supplemented on December 15, 1995, February 5, February 9, February 28, March 4, March 28 and April 3, 1996.

These amendments revise the LaSalle Technical Specifications (TSs) to reflect the deletion of the leakage control system (LCS) presently installed to control and contain the leakage past the main steamline isolation valves (MSIVs) on each of the four main steamlines. The TSs are also revised to raise the MSIV allowable leakage rates in TS Section 4.6.3.6a from 25 standard cubic feet per hour (scfh) for each steamline (a total allowable leakage of 100 scfh from all four main steamlines past the MSIVs) to values of 100 scfh per steamline (400 scfh for all four steamlines) when the MSIVs are tested at a pressure equal to or greater than 25 pounds per square inch gauge (psig). The applicable bases sections in the TSs are also revised.

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

Sincerely,

Original signed by:

M. David Lynch, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-373, 50-374

Enclosures:
1. Amendment No. 112 to NPF-11
2. Amendment No. 97 to NPF-18
3. Safety Evaluation

cc w/encl: see next page

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D. L. Farrar
Commonwealth Edison Company

LaSalle County Station
Unit Nos. 1 and 2

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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. 112
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 August 28, 1995, as supplemented on December 15, 1995, February 5, February 9, February 28, March 4, March 28 and April 3, 1996, 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), 2.D.(e), and 2.D.(f) of the Facility Operating License No. NPF-11* are hereby amended to read as follows:

---

*Pages 16a and 16b are provided, 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. 112, 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.(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. 102 to this license.

2.D.(f) An exemption was granted to remove the Main Steam Isolation Valves (MSIVs) from the acceptance criteria for the combined local leak rate test (Type B and C), as defined in the regulations of 10 CFR Part 50, Appendix J, Option B, Paragraph III.B. Exemption (f) is described in the safety evaluation accompanying Amendment No. 112 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 prior to startup from refueling outage L1R07.

FOR THE NUCLEAR REGULATORY COMMISSION

M. David Lynch, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachments:
1. License pages 16a and 16b
2. Changes to the Technical Specifications

Date of Issuance: April 5, 1996

2.C.(34) Deleted.

2.C.(35) <u>Surveillance Interval Extension</u>

The performance interval for those surveillance requirements identified in the licensee's request for surveillance interval extension dated April 11, 1995, shall be extended to April 5, 1996, to coincide with the Unit 1 seventh refueling outage schedule. The extended interval shall not exceed a total of 25.1 months for 18 month surveillances.

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. 102 to this license.

(f) An exemption was granted to remove the Main Steam Isolation Valves (MSIVs) from the acceptance criteria for the combined local leak rate test (Type B and C), as defined in the regulations of 10 CFR Part 50, Appendix J, Option B, Paragraph III.B. Exemption (f) is described in the safety evaluation accompanying Amendment No. 112 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 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. 112

FACILITY OPERATING LICENSE NO. NPF-11

DOCKET NO. 50-373

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

REMOVE	INSERT
VII	VII
XIV	XIV
3/4 6-7	3/4 6-7
3/4 6-8	-
3/4 6-23	3/4 6-23
3/4 8-30	3/4 8-30
B 3/4 6-2	B 3/4 6-2
B 3/4 6-4a	B 3/4 6-4a

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

3/4.6.1.4 and 3/4.6.1.5 INTENTIONALLY LEFT BLANK

Pages 3/4 6-8 through 3/4 6-12 DELETED

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 the frequency specified by the Primary Containment Leakage Rate Testing Program:

a. Verify leakage rate for any one main steamline through the isolation valves is $\leq$ 100 scfh, not to exceed 400 scfh for all four main steamlines, when tested at $\geq$ 25.0 psig.

b. Verify combined leakage rate through hydrostatically tested lines that penetrate the primary containment is within limits.

TABLE 3.8.3.3-1 (Continued)
MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

	VALVE NUMBER	BYPASS DEVICE (Continuous)(Accident Conditions)	SYSTEM(S) AFFECTED
l.	DELETED		
m.	1E22 - F004	Accident Conditions	HPCS system
	1E22 - F012	Accident Conditions	
	1E22 - F015	Continuous	
	1E22 - F023	Accident Conditions	

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.4 DELETED

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 than 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 steamline break accident from the full open position.

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 access 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 any one main steamline is $\leq$ 100 scfh, not to exceed 400 scfh for all four main steamlines, 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 steamlines through the isolation valves must be verified to be in accordance with the Primary Containment Leakage Rate Testing Program. 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 steamlines through the isolation valves is properly accounted for in determining the overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

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 in accordance with the Primary Containment Leakage Rate Testing Program.

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. 97
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 August 28, 1995, as supplemented on December 15, 1995, February 5, February 9, February 28, March 4, March 28 and April 3, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment and paragraph 2.C.(2) and 2.D.(e) of the Facility Operating License No. NPF-18* are hereby amended to read as follows:

---

*Pages 10 and 11 are provided, for convenience, for the composite license to reflect this change.

2.C.(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 97 , 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.(e) An exemption was granted to remove the Main Steam Isolation Valves (MSIVs) from the acceptance criteria for the combined local leak rate test (Type B and C), as defined in the regulations of 10 CFR Part 50, Appendix J, Option B, Paragraph III.B. Exemption (e) is described in the safety evaluation accompanying Amendment No. 97 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 prior to startup from refueling outage L2R07.

FOR THE NUCLEAR REGULATORY COMMISSION

M. David Lynch, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachments:
1. License pages 10 and 11
2. Changes to the Technical Specifications

Date of Issuance: April 5, 1996

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. 87 to this license.

(e) An exemption was granted to remove the Main Steam Isolation Valves (MSIVs) from the acceptance criteria for the combined local leak rate test (Type B and C), as defined in the regulations of 10 CFR Part 50, Appendix J, Option B, Paragraph III.B. Exemption (e) is described in the safety evaluation accompanying Amendment No. 97 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. 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.

I. This license is effective as of the date of issuance and shall expire at Midnight on December 16, 2023.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:
Darrell G. Eisenhut for

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachments/Appendices:
1. Attachment 1
2. Attachment 2
3. Appendix A - Technical Specifications (NUREG-1013)
4. Appendix B - Environmental Protection Plan

Date of Issuance: December 16, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 97

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
VII	VII
XIV	XIV
3/4 6-7	3/4 6-7
3/4 6-8	-
3/4 6-26	3/4 6-26
3/4 8-29	3/4 8-29
3/4 8-30	3/4 8-30
B 3/4 6-2	B 3/4 6-2
B 3/4 6-2a	B 3/4 6-2a
B 3/4 6-4a	B 3/4 6-4a

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

3/4.6.1.4 and 3/4.6.1.5 INTENTIONALLY LEFT BLANK

Pages 3/4 6-8 through 3/4 6-15 DELETED

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 the frequency specified by the Primary Containment Leakage Rate Testing Program:

a. Verify leakage rate for any one main steamline through the isolation valves is $\leq$ 100 scfh, not to exceed 400 scfh for all four main steamlines when tested at $\geq$ 25.0 psig.

b. Verify combined leakage rate through hydrostatically tested lines that penetrate the primary containment is within limits.

TABLE 3.8.3.3-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

	VALVE NUMBER	BYPASS DEVICE (Continuous)(Accident Conditions)	SYSTEM(S) AFFECTED
	2E12 - F099B	Accident Conditions	
	2E12 - F099A	Accident Conditions	
	2E12 - F008	Accident Conditions	
	2E12 - F009	Accident Conditions	
	2E12 - F040A	Accident Conditions	
	2E12 - F040B	Accident Conditions	
	2E12 - F049A	Accident Conditions	
	2E12 - F049B	Accident Conditions	
	2E12 - F053A	Accident Conditions	
	2E12 - F053B	Accident Conditions	
	2E12 - F006A	Continuous	
	2E12 - F023	Accident Conditions	
	2E12 - F027B	Accident Conditions	
	2E12 - F042A	Accident Conditions	
	2E12 - F042C	Accident Conditions	
	2E12 - F064C	Accident Conditions	
	2E12 - F094	Continuous	
k.	2E51 - F086	Accident Conditions	RCIC system
	2E51 - F022	Accident Conditions	
	2E51 - F068	Continuous	
	2E51 - F069	Continuous	
	2E51 - F080	Accident Conditions	
	2E51 - F046	Accident Conditions	
	2E51 - F059	Accident Conditions	
	2E51 - F063	Accident Conditions	
	2E51 - F019	Accident Conditions	
	2E51 - F031	Continuous	
	2E51 - F045	Accident Conditions	
	2E51 - F008	Accident Conditions	
	2E51 - F010	Accident Conditions	
	2E51 - F013	Accident Conditions	
	2E51 - F064	Accident Conditions	
	2E51 - F076	Accident Conditions	
l.	DELETED		
m.	2E22 - F004	Accident Conditions	HPCS system
	2E22 - F012	Accident Conditions	
	2E22 - F015	Continuous	
	2E22 - F023	Accident Conditions	

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3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY (Continued)

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 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.

Maintaining primary containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Primary Containment Leakage Rate Testing Program. The surveillance requirements reflect the leakage rate testing requirements with respect to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and primary containment OPERABILITY testing. The periodic testing requirements verify that air lock leakage does not exceed the allowed fraction of the overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program. Additional annotation is provided to require the results of air lock leakage tests being evaluated against the acceptance criteria applicable to the surveillance requirements. This ensures that the air lock leakage is properly accounted for in determined the combined Type B and Type C primary containment leakage.

3/4.6.1.4 DELETED

3/4.6.1 PRIMARY CONTAINMENT

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 than 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 steamline break accident from the full open position.

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 access 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 any one main steamline is $\leq$ 100 scfh, not to exceed 400 scfh for all four main steamlines 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 steamlines through the isolation valves must be verified to be in accordance with the Primary Containment Leakage Rate Testing Program. 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 steamlines through the isolation valves is properly accounted for in determining the overall primary containment leakage rate. The frequency is required by the Primary Containment Leakage Rate Testing Program.

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 in accordance with the Primary Containment Leakage Rate Testing Program.

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. 112 TO FACILITY OPERATING LICENSE NO. NPF-11 AND

AMENDMENT NO. 97 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

In its application dated August 28, 1995, as supplemented by information in letters dated December 15, 1995, February 5, February 9, February 28, March 4, March 28 and April 3, 1996, Commonwealth Edison Company (ComEd, the licensee) proposed license amendments which would revise the Technical Specifications (TS) of LaSalle County Station, Units 1 and 2. These revisions would increase the allowable main steam isolation valve (MSIV) leakage rate and delete the MSIV leakage control system (LCS). ComEd proposed to use the main steam piping, drain lines, and main condenser as an alternate leakage treatment (ALT) pathway for MSIV leakage.

The submittals dated December 15, 1995, February 5, February 9, February 28, March 4, March 28 and April 3, 1996, contained only clarifying information and did not change the scope of the application or the initial proposed no significant hazards consideration determination.

The application proposed to revise License Condition 2.D(f) of Unit 1 (NPF-11) and License Condition 2.D(e) of Unit 2 (NPF-18) to reflect a modification to exemptions previously granted regarding certain leakage test requirements in Appendix J to 10 CFR Part 50. Additionally, the licensee proposed to delete TS Sections 3/4.6.1.4 and 3/4.6.1.5 in the Unit 1 and 2 TSs to reflect the proposed deletion of the LCS. The application also proposed to increase the TS allowable MSIV leakage values in TS Section 4.6.3.6.a for both units from 25 standard cubic feet per hour (scfh) per main steamline and a maximum value of 100 scfh for all four main steamlines to a value equal to or less than 100 scfh for each main steamline, not to exceed 400 scfh for all four main steamlines. Finally, the licensee proposed to delete the list of motor operated valves associated with the functioning of the LCS from Table 3.8.3.3-1 in the TSs of both units. The appropriate index pages and TS bases sections would also be revised to reflect the changes cited above.

By letter dated March 4, 1996, the licensee revised its request for exemption which had been included in its August 28, 1995, request for license amendments. In the August 28 letter, the licensee had requested an exemption from Type A tests. However, no type A test exemption is required because the licensee has implemented the new performance based Containment Leakage Testing Rule, 10 CFR Part 50, Appendix J, Option B. This was approved by the Commission by letter dated March 11, 1996. Option B separates local leak rate tests, Type B and C, from the acceptance criteria for a Type A test. Therefore, the revised license pages included with this amendment do not include the exemption from Type A tests as originally requested by the licensee. In addition, the paragraph discussing the exemption from Type B and C tests was revised to reflect 10 CFR Part 50, Appendix J, Option B, Paragraph III.B.

These proposed changes to the TS are related to research performed by the Boiling Water Reactor Owners' Group (BWROG), as documented in the General Electric Company (GE) Report, NEDC-31858P, Revision 2, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," dated September 1993. However, ComEd chose in its proposal not to utilize the BWROG earthquake experience database to qualify piping and supports in the ALT pathway. ComEd relies, instead, on plant-specific analytical evaluations for the seismic adequacy of the MSIV ALT system piping, structures and components, supplemented by a plant seismic walkdown. Accordingly, the staff's review of the proposed TS changes was based entirely on the analytical evaluation of the seismic adequacy of the ALT system piping, components and the corresponding supports. In addition, since the turbine building houses the majority of the ALT system piping and components, its seismic adequacy was also assessed.

The staff's acceptance of the licensee's proposals to delete the LCS and revise the TS allowable MSIV leakage is based on determining whether the radiological consequences of the proposed TS revisions and the hardware modifications are acceptable. Specifically, the radiation exposures must meet the requirements in General Design Criterion (GDC) 19 of Appendix A to 10 CFR Part 50 regarding radiation doses to control room personnel and must not exceed the guideline values for radiation exposures to the public in Section 100.11 of 10 CFR Part 100. Accordingly, the staff performed an independent evaluation of the radiological consequences of the revisions proposed by the licensee.

2.0 DISCUSSION

2.1 Introduction

Each of the four main steamlines have a set of quick-acting MSIVs which close in the event of a severe transient or accident. One of the MSIVs is inboard of the primary containment structure and the other is outboard of the primary containment. The main design requirements of these boiling water reactor (BWR) MSIVs are that they close within 5 seconds against the full power steam flow and that they be tested periodically during a fuel cycle. Historically, these BWR MSIVs have had relatively large leakage rates which represent for the LaSalle Station, over 40 percent of the total allowable leakage, with margin, from the primary containment (i.e., 0.6 L_a).

Accordingly, this leakage past the MSIVs was controlled and processed to minimize the radiological consequences of this leakage in the event of a loss-of-coolant accident (LOCA). This led to the installation of the LCS in the LaSalle Station. The evaluation of this system and its mitigation of the radiological consequences of MSIV leakage under LOCA conditions is presented in the staff's Safety Evaluation Report (SER) (NUREG-0519, dated March 1981), as further evaluated in Supplement No. 6 to the SER issued in November 1983.

The LaSalle Station, which originally operated for a 12-month fuel cycle, is presently licensed to operate for an 18-month fuel cycle and will soon start operating on a 24-month fuel cycle. As the length of the fuel cycle is expanded, there is a potential for the measured MSIV leakage at the end of cycle (EOC) to increase to amounts greater than that presently experienced. Furthermore, the capacity of the LCS is limited to 100 scfh.

In light of the potential for MSIV leakage to exceed the TS allowable MSIV leakage rates for operating BWRs and the limited capacity of the LCS, the BWROG conducted research into design features which could serve as alternate treatment paths for the present LCS. This effort culminated in the GE Report cited in Section 1.0, above. ComEd's pending proposal for license amendments, cited above, represents a plant-specific application for LaSalle of the industry generic approach to resolve this issue of processing and controlling MSIV leakage.

2.2 Description of the ALT Pathway

The primary components to be relied upon for the proposed ALT system are the main turbine condenser and the primary drain pathway piping. Leakage past the outboard MSIVs travels down the four 26-inch main steamlines to either the upstream drain line designated as Primary ALT Path A or to the downstream drain line designated as Primary ALT Path B, into the main condenser. Each of these ALT leakage paths consists of the following:

1. Four main steamlines from their respective MSIVs to their respective drain lines.

2. A 2-inch drain line connected to each steamline.

3. A 12-inch drain header, receiving MSIV leakage from each of the four 2-inch drain lines.

4. A 3-inch line is routed from the 12-inch drain header and branches into the 1-inch normal operating orifice drain line and the 3-inch startup drain line as described below:

    a) An operating 1-inch drain line with an 0.875-inch orifice connected to the condenser at elevation 696'-7" and a normally open motor-operated globe valve. The bottom of the condenser is at elevation 690'-7".

b) A startup 3-inch drain line without an orifice connected to the condenser at elevation 696'-7" and a normally closed motor-operated globe valve.

During normal operation, the operating drain valves are open and the startup drains are closed. For the ALT mode of operation, the two operating drain valves remain open and either one of the two startup drain valves is opened. This assures an initial flow path, although restricted, until a startup drain valve can be opened. No credit is taken in the staff's radiological dose estimate for the two operating drain lines being open.

The condenser forms the ultimate boundary of the ALT pathway. Boundaries upstream of the condenser were established by utilizing existing valves which thereby defined the extent of the ALT pathway subject to seismic review. The criteria used to define the components which limit the scope of the seismic review are:

1. Normally closed valves which will not open and can be assured to remain closed.

2. Normally open valves which can be assured to close and remain closed.

3. Valves which may require operator action to assure closure and are powered from a reliable power source.

4. Drain lines connected to the main condenser which will be utilized to carry the MSIV leakage to the condenser.

2.3 <u>Radiological Dose Assessment</u>

In order to demonstrate the adequacy of the engineered safety features (ESFs) designed to mitigate the radiological consequences of the design basis accidents (DBAs) with a maximum TS allowable MSIV leak rate of 400 scfh total from all four main steamlines, the licensee assessed the offsite and control room radiological consequences which could result from the occurrence of a postulated LOCA and presented the results of that assessment in their submittal dated August 28, 1995. The staff previously assessed the offsite radiological consequences of a LOCA with MSIV leakage increased from 11.5 to 25 scfh in Supplement No. 6 to the LaSalle SER. In this supplement, the staff considered the current 100 scfh MSIV total leak rate from four main steamlines in the main steamline isolation valve leakage transport path (i.e., the LCS) to the environment following a postulated LOCA.

In its independent evaluation of the radiological consequences of the licensee's proposal, the staff recalculated the radiation doses associated with the proposed ALT MSIV leakage path assuming that the radiological consequences associated with the other radioactivity transport paths would be negligibly affected by the proposed amendments. Accordingly, the radiological consequences of these other pathways were not recalculated.

The procedures used in the staff's calculation of the radiological consequences associated with MSIV valve leakage were based upon: (1) the TID-14844 source term, consistent with the guidelines provided in the applicable sections of the Standard Review Plan (SRP) (NUREG-0800) and the appropriate Regulatory Guides; and (2) the assumptions and parameters used in the staff's SER cited above, except for the following three deviations. The staff has accepted credit for radioactive iodine removal in the main steamlines, drain lines and main condenser by hold-up, decay and deposition. Dose contributions to the whole body from the increased MSIV leakage were recalculated based upon the ratio of the proposed TS leakage rate limit of 400 scfh to the current TS limit of 100 scfh. No credit was given for holdup and decay of noble gases in the main steamlines and condenser. In addition, the staff calculated the relative concentration for the control room assessment based on the size of the building wake cavity needed to capture the postulated effluent release rather than on the minimum building cross-sectional area.

The current assumption used by the staff in calculating radiological consequences of potential DBAs for operating plants is based upon a conservative assumption that the leakage limit allowed by a plant's TS is released directly into the environment. No credit is currently taken for the integrity and leak tightness of the main steam piping and condenser to provide holdup and plateout of fission products.

3.0 EVALUATION

3.1 Structural Evaluation

3.1.1 Reliability and Structural Integrity of the ALT Path, Including Boundary Valves

As ComEd stated in its letter dated February 5, 1996, the ALT pathway has high reliability because the LaSalle Station will have redundant, seismically qualified ALT paths to the main condenser. Accordingly, mechanical failure of a single valve in one ALT drain path does not prevent routing MSIV leakage through the redundant path to the condenser. Even in the remote chance of failure of all three power sources (i.e., two offsite power sources and the safety-related diesel generator), a restricted flow path through the operating drain orifices will still direct the MSIV leakage to the main condenser.

The licensee also indicated in the same letter that the highly reliable boundary isolation valves fall into the following four categories:

1. All seven (per unit) of the remote manual motor-operated valves and motor operators were originally seismically qualified. While they were subsequently reclassified as non-safety related, they are powered from their original reliable power sources which are the ESS Division 2 busses.

2. Local manual valves used as boundary valves are seismically qualified and remain in their normal operating (i.e., closed) positions and require no operator action.

3. The main steam high pressure (HP) turbine main stop valves are operated utilizing electrical hydraulic control (EHC) pressure and fail closed upon either loss of electrical power to EHC, loss of EHC pressure, or upon a turbine trip. The main steam bypass valves are also operated utilizing EHC pressure and fail closed upon loss of electrical power to EHC or loss of EHC pressure. These valves have been evaluated and determined to be seismically rugged.

4. The dual-acting, quick-closing MSIVs are safety-related valves and are seismically qualified.

The licensee has stated that all the motor-operated valves utilized as either boundary valves or ALT path control valves will be included in the LaSalle inservice test (IST) program, and will be stroke tested once per fuel cycle. This commitment is acceptable.

In addition, the piping and pipe supports in the ALT pathway are highly reliable because the piping and its supports within the ALT boundary will have been seismically qualified prior to startup, as described in the following sections.

3.1.2 <u>Seismic Walkdowns</u>

A plant walkdown was performed by the licensee in accordance with Sargent & Lundy's walkdown criteria provided in its report, EMD-067927, Revision 0. The walkdown was focused on visually identifying conditions of piping and support configurations which may result in seismically-induced pressure boundary failure and inventory release from the main steam and drain piping. The potential vulnerabilities which are identified as "outliers" may include: (1) failure of non-seismically designed piping (i.e., the class D portion of the subject piping); (2) failure of poor installations and deterioration of pipe supports, collapse of non-seismically designed plant features which may impact the seismically designed systems (II/I); (3) seismic interactions; and (4) differential seismic motions on piping systems. All outliers identified during the walkdown of the proposed ALT pathway for LaSalle were evaluated by review of the existing analyses or design drawings. As stated in ComEd's letter dated August 28, 1995, the two outliers requiring further actions are:

1. For the process sampling line which has no automatic or powered isolation valve to isolate its leak path, the licensee stated that it would either: (1) evaluate the radiological effect of the unisolated MSIV leakage path, (2) install automatic/reliable powered isolation; or (3) administratively control the manual isolation valve closed.

2. Verify the seismic adequacy of the concrete block wall supporting the pressure sensing instrument lines and the concrete block walls which are

close to the pressure sensors. In addition, the licensee would provide methods of reinforcement or isolation for these components if they are found to be seismically inadequate.

In the letter dated March 28, 1996, the licensee stated that it had chosen to use an administrative control for resolving the process sampling line outlier described above in Item 1. The staff finds this acceptable. The resolution of the second outlier issue is discussed in Section 3.1.6 of this safety evaluation (SE).

3.1.3 Structural Analyses of ALT Pathway and Condenser Structural Components

3.1.3.1 Analyses of ALT Pathway

As stated in ComEd's letters dated August 28 and December 15, 1995, the affected piping (except the pressure sensing lines and the Unit 2 main steam downstream drain line subsystem) have been seismically analyzed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Section III, Class 2 and 3 rules, using response spectrum analysis techniques for the operating basis earthquake (OBE) and safe shutdown earthquake (SSE) loads. The corresponding pipe supports were also designed for these seismic loads using the ANSI B31.1 Code for hardware design and applicable AISC allowables for auxiliary steel design. The expansion anchor assemblies for the pipe supports were designed in accordance with the criteria contained in NRC IE Bulletin 79-02, as documented in the licensee's submittal dated March 15, 1982, "Final Report on Pipe Support Base Plate Design Using Concrete Expansion Anchor Bolts." Anchorages designed in accordance with IE Bulletin 79-02 are acceptable.

All of the piping systems within the scope of the ALT pathway seismic review are classified as non-safety-related, although a majority of them were seismically analyzed (Class D+) in accordance with ASME Section III, Class 2 and 3 rules. The seismically analyzed piping includes the main steamline (i.e., the downstream piping from the outboard MSIVs to the main steam stop valves, the main steam bypass valves and main steam auxiliary supply steam stop valves), drain lines from main steam piping to the condenser, and the warm-up lines to Valve 1B21-F020. Small bore instrument lines such as process sampling lines have also been seismically designed using a simplified procedure to support the analysis of piping/tubing. The design methods for all these lines are consistent with Seismic Category I qualification methods. The design margins using these methods are acceptable, thereby, ensuring good seismic performance.

One model for this analysis included the main steam piping to the turbine and the bypass line. The main steam drain and warm-up lines were decoupled from the main steamline and were analyzed up to the condenser and structural anchors, respectively. These piping subsystems consist of the majority of the piping and supports within the scope of the seismic review of the ALT pathway and the design methods for these lines are consistent with Seismic Category I qualification methods for LaSalle's safety-related piping and supports. On this basis, we find the analyses of these piping subsystems acceptable.

As discussed above, the only pipe lines which were not seismically designed are the pressure sensing lines (seven for each unit) and the Unit 2 main steam downstream drain line subsystem 2MS-71. The licensee has since performed seismic analyses for the pressure sensing lines in response to the staff's request for additional information and summarized these analyses and the results in its letter dated February 5, 1996.

For Unit 1, four of the pressure sensing lines are composed of 1-inch and 1/2-inch piping, and are connected to the main steam header near the main steam HP turbine main stop valves. Two of the pressure sensing lines are composed of 1-inch and 1/2-inch piping and 3/8-inch stainless tubing and are connected to the main steamline. The last pressure sensing line is composed of 3/4-inch and 1/2-inch piping, and 3/8-inch and 1/4-inch stainless steel tubing and is connected to the main steam pressure equalizing header. The loading considered in the licensee's analyses included dead weight, thermal, pressure and seismic loads utilizing the envelopes of the turbine building wall and slab response spectral acceleration curves.

The licensee stated in its letter dated February 5, 1996, that the piping analyses cited above have been completed in accordance with the LaSalle Updated Final Safety Analysis Report (UFSAR) requirements. The results indicated that the pipe stresses are within the design allowable stress limits, and adequate safety margins also exist for the associated pipe supports.

The licensee also stated that the pressure sensing lines for Unit 2 will also be similarly analyzed, and pipe supports modified, if required, prior to startup from the forthcoming refueling outage for Unit 2 in which the MSIV-LCS is eliminated from service.

Based on the foregoing considerations, the staff determined that the licensee's approach to ensure the seismic adequacy for the pressure sensing lines is acceptable. For the Unit 2 main steam downstream drain line subsystem 2MS-71, the licensee will confirm that similar seismic analyses will be performed and the pressure sensing lines have been demonstrated to be acceptable prior to the startup of Unit 2 from its forthcoming refueling outage.

3.1.3.2 Structural Analyses of the Main Condenser

The LaSalle main condenser is a single shell, single pass, deaerating type condenser with a divided water box constructed in accordance with the Heat Exchange Institute (HEI) standards. The overall dimensions of the condenser are 70 feet high, 35 feet wide, and 90 feet long.

The normal operating pressure in the steam compartment is between about 0.5 pounds per square inch absolute (psia) and 2.5 psia. The inlet and outlet water boxes, condenser tubes, and wet well at the base of the condenser are full of water during normal operation. The 7/8-inch thick shell of the condenser is stiffened by the tube support plates interconnected by struts that connect the support plates to the side walls and condenser bottom. These support plates are spaced about 40 inches along the length of the tubes.

The condenser is seated on eight concrete piers, arranged in a symmetrical fashion about the condenser's longitudinal and transverse dimensions, which are supported by the turbine building foundation. The four interior condenser piers are 6 feet by 8 feet -10 inches in cross-section and are integral with the substantially larger turbine pedestal piers. The four corner piers are of the same size and are also integral with the larger adjacent turbine pedestal piers. Each support uses six 1-5/8-inch diameter A36 bolts to anchor the condenser to the piers. One of the interior supports acts as the stationary anchor point while the other seven are sliding supports used to accommodate the condenser's thermal movement using oversized slotted bolt holes in the base plates.

The condenser was hydro-tested by filling the shell with water to a level 2 feet above the turbine isolation expansion joints. This hydrostatic test loading condition applies twice the operating weight to the condenser base and support pier than is present during normal operating conditions. The loads on the condenser support pedestals from this hydro-test exceed the reactions from operating loads plus vertical seismic and overturning moments by 70 percent. This load test demonstrates the condenser's ability to adequately resist the vertical loads of the SSE.

The seismic loads in the north-south (N-S) direction are resisted by the connections at the condenser base through the axial stiffness of the longitudinal shell plates. The shell is constructed from 7/8-inch thick ASTM A285, Grade C, flange quality steel, and is laterally braced every 40 inches by struts used to support the tube support sheets. The shell side walls experience a maximum shear stress of less than 2 kips per square inch (ksi) from the N-S seismic force, which is relatively insignificant.

The effect of east-west (E-W) seismic loads on the local load carrying capacity of the condenser shell is also small in comparison with the hydrostatic test load. The water pressure at the top of the steam compartment walls during the hydrostatic test was 11 psi and increased to 28 psi at the base of the condenser. The equivalent lateral seismic load that the tubes would apply on the side walls is less than 4 psi. Similarly, the lateral pressure from water in the hot well will be less than 2 psi. Comparison of these equivalent design pressures demonstrates that there is substantial design margin for the E-W seismic loads from the condenser tubes and hot well.

The loads associated with the heaters and the water boxes, however, act like concentrated loads and are carried to the E-W support points through the condenser acting as a girder. In this regard, a simple representation of the stresses induced by E-W seismic loads is to treat the condenser itself as a 35 foot deep girder, with both ends cantilevering past the interior supports.

The resulting bending moment causes flexural stresses in the side plates, acting as the flanges of the beam, of less than 1.5 ksi. Finally, the stiffness required to resist the E-W seismic loads at the interior support points is provided by the interior tube support plates and their support brackets. These large steel plates and internal support components have been assessed and found to be within allowable stresses under seismic loads.

Based on the foregoing considerations, the licensee determined, and the staff agrees, that the condenser shell and internal components are seismically rugged and is capable of transferring SSE forces to the supporting structure. On this basis, we find that the use of the main steam condenser as part of the ALT pathway is acceptable.

3.1.4 Bounding Seismic Analyses

The licensee provided some original design documents for two subsystems, 2MS-31B and 2MS-56, in its letter dated December 15, 1995. Additional information on the analysis of the two subsystems was subsequently provided by the licensee in its letter dated February 5, 1996. The licensee stated that the two subsystems were selected at random from the total population of affected subsystems that had been originally seismically analyzed in accordance with the UFSAR.

Subsystem 2MS-31B is the warm-up by-pass line to the main steamlines downstream of the MSIVs, consisting of a combination of Schedule 80 pipes ranging from 3/4 inch to 12 inches in diameter. Subsystem 2MS-56 is the upstream drain header from the main steamlines to the condenser (which is an ALT flow path to the condenser), consisting of a combination of Schedule 80 pipes ranging from 1 inch to 12 inches in diameter. The pipes are designated as Class D, except for a pipe between Penetration M-22 and Valve 2B21-F019 which is Class A. The loadings considered in the licensee's analyses were dead weight, thermal, pressure and the response spectral accelerations which represent the envelope of the effects of design earthquakes and safety relief valve (SRV) loadings.

The licensee's seismic analysis methodology for the ALT pathway is in accordance with the LaSalle licensing commitments as delineated in the LaSalle UFSAR. Specifically, the damping values used are 1/2 percent for the OBE/SRV loads and 1 percent for the faulted conditions for subsystem 2MS-31B, and 1/2 percent for both the OBE and SSE for subsystem 2MS-56. An absolute sum method is used for modal response combinations and the cut-off frequency is above 33 Hz. The square root of the sum of the squares (SRSS) method is used for seismic directional response combinations. Loading combinations for pipe stresses and support loads are as delineated in Table 3.9-16 of the LaSalle UFSAR for each Service Level. In addition, the pipe stress allowable limits are in accordance with ASME Code Section III, 1974 Edition. The load capacities for pipe supports are in accordance with the allowables recommended by the vender for standard components and the AISC Manual for auxiliary steel design.

The seismic analysis was performed using Sargent & Lundy's piping analysis program PIPSYS, which has been bench-marked by the licensee to the requirements of the NRC SRP and has been used extensively by industry. The highest seismic stresses in both subsystems are found to be significantly below the corresponding allowable stress limits. The piping support designs also meet the UFSAR structural limits with adequate margins. The licensee, therefore, concluded, and the staff agrees, that the two piping subsystems chosen as representative of the ALT system piping are seismically rugged and are, therefore, acceptable for use in the ALT pathway.

3.1.5 Analyses of Turbine Building

In its letter dated December 15, 1995, the licensee stated that the roof of the turbine building had been seismically designed using the 1970 Uniform Building Code (UBC) and the shear walls and slab of the turbine building were dynamically analyzed and designed for an SSE. The licensee also stated that the design of the roof steel structure was governed by the tornado wind load, instead of the UBC seismic load.

In its submittal dated February 28, 1996, the licensee stated that it had performed a response spectrum analysis for the roof structure of the turbine building and calculated a maximum shear force of 1174 kips in the east-west direction and 1026 kips in the north-south direction. These shear forces are well below 2694 kips and 3438 kips, respectively, for which the roof structure was originally designed under the tornado wind load. On March 13, 1996, the licensee transmitted to the staff, the corresponding safety margin calculations for the roof structure design under the vertical seismic load. The maximum tensile stress for each roof steel girder of the turbine building was calculated based on the assumption that it was simply supported between columns. The allowable tensile stresses for the roof girders under the SSE loads were obtained by multiplying the allowable stresses by the factor of 1.6 recommended in the AISC Manual of Steel Construction, and are shown to be less than 95 percent of the material yield stress. Safety margins for the roof girders were then obtained by dividing the allowable tensile stresses by the maximum tensile stresses generated under the vertical SSE loads. The results indicated a minimum safety margin of 2.6. The staff found the licensee's methodologies for calculating the maximum and allowable tensile stresses of the roof girders to be acceptable. The staff, therefore, finds that the roof structure of the turbine building has acceptable design margins of safety for the SSE loading condition.

3.1.6 Adequacy of Masonry Wall Design

The licensee stated in its letter dated August 28, 1995, that pressure sensing lines 1MS93AA/AC/AD-1, 1MS68AB/BB-1 and 1MS69AB-1/2 penetrated a concrete block wall, and that valves and pressure sensors are mounted on the other side of this concrete block wall. Additionally, there exist other block walls which are located near the pressure sensors; their failure could potentially impact the operability of these sensors. The licensee stated in its letter dated April 3, 1996, that it intends to physically modify these subject walls or strengthen them to the criteria of NRC IE Bulletin 80-11 for masonry wall design prior to restart of the units during the refueling outage in which the LCS is deleted. The staff finds this acceptable.

3.1.7 Adequacy of the Main Steam Condenser Anchorage

In its letter dated February 5, 1996, the licensee stated that it would use structural steel members to fill the gap between the condenser wall and the turbine pedestals in the east-west direction to provide seismic restraint for the condenser. Therefore, the anchor bolts will experience no shear load in the east-west direction during earthquakes because the steel fillers would absorb all seismic loads. In its letter dated February 28, 1996, the licensee stated that it had designed a support system for the condenser which can resist a seismic load of 2400 kips in the north-south direction which is well in excess of 1862 kips seismic shear load, in addition to the anchor bolt shear capacity of 1625 kips. The staff reviewed the method used by the licensee to calculate the shear resistance of the condenser anchorage and finds it acceptable. The licensee also stated in its letter dated February 28, 1996, that the required support modifications would be completed for each unit prior to startup from the refueling outage in which the LCS is deleted. The staff finds this commitment acceptable.

3.2 Radiological Assessment

3.2.1 Iodine Transport and Deposition Models

The radioactive releases postulated to be released by TID-14844 in the event of a LOCA includes radioiodine which is a principal contributor to the radiation doses both onsite and offsite. Accordingly, in evaluating the licensee's amendment requests, it is necessary to evaluate the iodine transport and depositions in the proposed ALT pathway. This section addresses these two processes.

Basic chemical and physical principles predict that gaseous iodine and airborne iodine particulate material will deposit on surfaces. Several laboratory and in-plant studies have demonstrated that gaseous iodine deposits by chemical adsorption and that particulate iodine deposits through a combination of sedimentation, molecular diffusion, turbulent diffusion, and impaction. Gaseous radioiodine exists in nuclear power plants in several forms: elemental (I_2), hypoiodous acid (HOI), organic (CH_3I), and particulate. In accordance with Regulatory Guide 1.3, the staff assumed 91 percent of the iodine released into the primary containment volume in the event of a LOCA is in the elemental form (including hypoiodous acid), 5 percent in the particulate form, and 4 percent in the form of organic iodides. It is further assumed that this iodine release is uniformly mixed in the primary containment and this mixture then leaks past the MSIVs at the TS allowable leakage rate.

Each of these forms of iodine deposits on surfaces at a different rate, described by a parameter known as the deposition velocity. The elemental iodine form, being the most reactive, has the largest deposition velocity and organic iodide has the smallest. Further, studies of in-plant airborne radioiodine show that iodine (elemental and particulate) deposited on the surface undergoes both physical and chemical changes and can either be resuspended as an airborne gas or become permanently fixed to the surface. The data also show that iodine can change its physical form so that iodine deposited as one form (usually elemental) can be resuspended in the same or in another form (usually organic). Conversion can be described in terms of resuspension rates that are different for each iodine species. Chemical surface fixation can similarly be described in terms of a surface fixation rate constant.

The transport of gaseous iodine in elemental and particulate forms has been studied for many years and several groups proposed different models to describe the observed phenomena (References 1 through 5). The staff used the model specifically developed by an NRC contractor (Reference 6) for iodine removal in BWR main steamlines and the main condenser following a LOCA.

The staff model treats the MSIV ALT pathway as a sequence of small segments for which instantaneous and homogeneous mixing is assumed. The mixing computed for each ALT segment is passed along as input to the next ALT segment. The number of ALT segments depends upon the parameters of the line and the flow rate and can be as many as 100,000 for a long, large-diameter pipe and a low flow rate. Each line segment is divided into five compartments that represent the concentrations of the three airborne iodine species, the surface that contains iodine available for resuspension, and surface iodine which has reacted and is fixed on the surface.

The staff's transport and deposition model considers three iodine species: elemental, particulate, and organic. A fourth species, hypoiodous acid, was considered for the purpose of the staff's model to be a form of elemental iodine. All radioiodine in an ALT segment undergoes radioactive decay. The resulting iodine concentration from each ALT segment of the deposition compartment serves as the input to the next ALT segment.

The GE model in the BWROG report cited above, as well as the one developed and used by the staff, is based on time-dependent temperature adsorption phenomena with instantaneous and perfect mixing in a given ALT volume. Both models use the same MSIV leakage pathways. However, they differ in the treatment of the buildup of iodine in the main steamlines and the condenser. The GE model assumed steady state iodine in equilibrium in a large volume while the staff model assumed transient buildup of iodine in a finite number of small volumes. The staff does not consider these differences to be significant since the resulting iodine deposition and removal rates in the main steamlines and condenser are in good agreement between both models.

The staff's transport model also assumed iodine transport through the condenser as a dilution flow rather than the plug flow in the steamlines. The staff assumed that the iodine input into the condenser mixes instantaneously with a volume of air in the condenser and that the diluted air exhausts at the same time and at the same rate as the input MSIV leakage flows into the condenser.

The staff developed the equations for iodine deposition velocities, resuspension rates, and surface fixation rates as a function of temperature using published data found in the literature. The equations and data are contained its contractor's report cited above. The equation for the deposition velocity of elemental iodine is based on a least-squares fit to the available data. Deposition velocity equations for HOI and organic iodine are based on the values at 30°C. Due to the lack of data at elevated temperatures, their temperature dependence is assumed to be similar to elemental iodine. Resuspension and fixation equations as a function of temperature are based on measurements available in the literature at ambient temperature. The staff assumed that resuspension and fixation rates will increase with increasing temperature.

The technical references and the GE and staff iodine deposition models indicate that particulate and elemental iodine would be expected to deposit on surfaces with rates of deposition varying with temperature, pressure, gas composition, surface material, and particulate size. Therefore, the staff believes that an appropriate credit for the removal of iodine in the main steamlines and main condensers is acceptable in the radiological consequence assessment following a design basis accident. This credit for the deposition of radioiodine in the ALT pathway components was factored into the staff's independent radiological assessment of the licensee's license amendment requests.

Sections III(c) and VI of Appendix A to 10 CFR Part 100, require that structures, systems, and components necessary to ensure a plant's capability to mitigate the radiological consequences of accidents which could result in radiation exposures comparable to the dose guidelines of 10 CFR Part 100, be designed to remain functional during and after an SSE. Thus, the main steamline, portions of the ALT pathway piping, and the main condenser are required to remain functional after an SSE if credit is taken for deposition of radioiodine. Consequently, the staff's practice has been to classify these components as safety-related and seismic Category I. In addition, Appendix A to 10 CFR Part 100 requires that the engineering method used to ensure that safety functions are maintained during and after an SSE involve the use of either a suitable dynamic analysis or a suitable qualification test. These requirements were evaluated in Section 3.1 of this SE and found to be acceptable.

Specifically, the staff determined that the ALT pathway will retain sufficient structural integrity to transport the relatively low MSIV leakage flow rate of about 2 to 3 feet per minute through the main steamlines to the condenser. The staff assumes in its radiological assessment that the condenser is open to the atmosphere via leakage through the low pressure turbine seals. Thus, it was only necessary to ensure that gross structural failure of the condenser will not occur under SSE conditions.

3.2.2 Control Room Habitability

The staff has previously evaluated the control room operator doses following a postulated LOCA and found that the calculated doses were within the guidelines of SRP Section 6.4. In this evaluation, the staff considered the fission product releases from the low pressure turbine seal due to the MSIV leakage up to the proposed limit of 400 scfh through the MSIV main steam drain lines and the main condenser. The staff reviewed the licensee's assessment, performed an independent evaluation of the atmospheric dispersion factors and found the values of the relative concentration estimates calculated by the licensee for the control room operator dose assessment, reasonably conservative. In its independent assessment, the staff assumed a ground level release of airborne fission products from the standby gas treatment system (SGTS) vent to the dual control room emergency air intakes. The SGTS vent is about twice as tall as the nearest solid adjacent structure, the reactor building. The size of the building wake cavity wherein the effluent is assumed to mix prior to entry into the control room intakes, is usually estimated by the projected minimum cross-sectional area of the buildings assumed to contribute to the formation of the wake. The evaluation performed by the licensee assumed a building cross-sectional area larger than that found acceptable by the staff. However, the staff assumed that the wake cavity extended to a height above the reactor building sufficient to capture the effluent release. Because of the building, release point and intake configuration at the LaSalle site, the staff's assessment resulted in relative concentration estimates about equal to those calculated by the licensee.

The resultant dose calculated to control room personnel as a consequence of the proposed increase in the TS allowable for MSIV leakage is 15 rem to the thyroid. When this dose is added to the doses previously calculated for other pathways, the staff finds that the recalculated whole-body and equivalent organ doses (i.e., the thyroid) are still within the guidelines of SRP Section 6.4 and, therefore, the staff's conclusions are not affected and remain the same. The staff's recalculated offsite and control room operator doses resulting from a postulated LOCA and the parameters and assumptions used in the staff's recalculation are provided in Tables 1 and 2 of this SE, respectively.

As shown in Table 1 of this SE, the onsite and offsite radiation doses from a postulated LOCA, evaluated with the proposed increased values of the TS allowable MSIV leakage, are within the appropriate acceptance criteria and are, therefore, acceptable. On this basis, the staff finds the radiological consequences of the subject license amendments acceptable.

3.3 Conclusion

For the reasons stated above, the staff finds the licensee's application to modify the LaSalle Station Technical Specifications to increase the allowable MSIV leakage rate and to delete the LCS is acceptable since potential offsite and control room doses to personnel remain within the limits of 10 CFR

Part 50, Appendix A; GDC 19; and 10 CFR Part 100; and are consistent with the guidance in SRP Section 6.4.

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 54717). 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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Table 1

<u>Radiological Consequences of Loss-of-Coolant Accident (rem)</u>

	EAB		LPZ	
	Thyroid	Whole Body	Thyroid	Whole Body
Bypass and MSIV Leakage	**142**	**3.0**	**14**	**0.3**
Containment Leakage	**113**	**3.0**	**10**	**0.3**
Total	**255**	**6.0**	**24**	**0.6**
10 CFR 100.11 Acceptance Criteria	300	25.0	300	25.0

	Thyroid	Whole Body
Control Room Operator Doses	30	4.0
Control Room GDC-19 Requirments	30	5.0

Table 2

Assumptions Used to Evaluate the MSIV Leakage Contribution

Core Thermal Power (MWt):	3458
MSIV Total Leak Rate: (100 scfh/MSIV)	400 scfh
Core Radionuclide Fractions Released to Drywell (%)	
Noble gases:	100
Iodines:	50
Forms of Iodine Species (%)	
Elemental:	91
Organic:	4
Particulate:	5
Iodine Dose Conversion Factors:	ICRP-30
Suppression Pool Decontamination Factor	
Noble gas	1
Organic iodine	1
Elemental iodine	10
Particulate	10
Containment Free Volume (ft^3):	4.73×10^5
Control Room Free Volume (ft^3):	1.17×10^5
Atmospheric Relative Concentrations (sec/m^3)	
0 - 1 hour, Exclusion Area Boundary:	9.1×10^{-5}
0 - 1 hour, Low Population Zone:	8.9×10^{-6}
1 - 2 hour, Exclusion Area Boundary:	2.5×10^{-6}
1 - 2 hour, Low Population Zone:	7.0×10^{-7}
2 - 8 hour, Low Population Zone:	7.0×10^{-7}
8 - 24 hour, Low Population Zone:	4.4×10^{-7}
1 - 4 day, Low Population Zone:	1.7×10^{-7}
4 - 30 day, Low Population Zone:	4.2×10^{-8}
Control Room Atmospheric Relative Concentrations (sec/m^3)	
0 - 8 hour:	2.65×10^{-4}
8 - 24 hour:	1.56×10^{-4}
1 - 4 day:	9.94×10^{-5}
4 - 30 day:	4.37×10^{-5}

7.0 REFERENCES

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