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:

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*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) Surveillance Interval Extension

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 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

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

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 ≤ 100 scfh, not to exceed 400 scfh for all four main steamlines, 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 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:

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*Pages 10 and 11 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. 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

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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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3/4.6.1 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.