Docket File
50-373/374

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

April 12, 1995

Mr. D. L. Farrar, Manager
Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2 - CORRECTION OF ISSUANCE OF AMENDMENTS (TAC NOS. M90702 AND M90703)

Dear Mr. Farrar:

On March 16, 1995, the U.S. Nuclear Regulatory Commission issued Amendment No. 102 to Facility Operating License No. NPF-11 and Amendment No. 87 to Facility Operating License No. NPF-18 for the LaSalle County Station, Units 1 and 2, respectively.

The pages listed below associated with those amendments were in error.

| Unit 1 | Unit 2 |
|---|---|
| Page 2, Amendment to License | Page 2, Amendment to License |
| Page 16b, Operating License | Page 10, Operating License |
| List of affected TS pages | List of affected TS pages |
| TS page 3/4 6-3 | TS page 1-3 |
| TS page 3/4 6-18 | TS page 3/4 3-13 |
| TS page 3/4 6-23 | TS page 3/4 6-3 |
| TS Bases page B 3/4 6-4 | TS page 3/4 6-21 |
| | TS Bases page B 3/4 6-4 |

Please substitute the enclosed corrected pages for the corresponding pages that were included in our March 16, 1995, transmittal.

170046



DF01

We regret any inconvenience this error may have created.

Sincerely,

Original signed by

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

Docket Nos. 50-373, 50-374

Enclosures: Corrected pages

cc w/encls: see next page

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D. L. Farrar
Commonwealth Edison Company

LaSalle County Station
Unit Nos. 1 and 2

cc:

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

The Technical Specifications contained in Appendix A, as revised through Amendment No. 102 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

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

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

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

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

(e) An exemption from the requirement of paragraph III.D of Appendix J to conduct the third Type A test of each ten-year service period when the plant is shutdown for the 10-year plant inservice inspections. Exemption (e) is described in the safety evaluation accompanying Amendment No. 102 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.

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. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, and the rules and regulations of the Commission (except as hereinafter exempted therefrom), and the provisions of the Act.

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

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

F. Reporting to the Commission:

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

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

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

ATTACHMENT TO LICENSE AMENDMENT NO. 102

FACILITY OPERATING LICENSE NO. NPF-11

DOCKET NO. 50-373

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain a vertical line indicating the area of change. The page marked with an asterisk is provided for convenience.

| REMOVE | INSERT |
|---|---|
| I | I |
| II | II |
| VII | VII |
| XXIII | XXIII |
| 1-3 | 1-3 |
| 1-4 | 1-4 |
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| 3/4 6-19 | 3/4 6-19 |
| 3/4 6-22 | 3/4 6-22 |
| 3/4 6-23 | 3/4 6-23 |
| 3/4 6-24 | 3/4 6-24 |
| 3/4 6-25 thru 6-34a | |
| 3/4 6-36 | 3/4 6-36 |
| 3/4 6-43 | 3/4 6-43 |
| *3/4 6-44 | *3/4 6-44 |
| B 3/4 6-1 | B 3/4 6-1 |
| | B 3/4 6-1a |
| B 3/4 6-4 | B 3/4 6-4 |
| | B 3/4 6-4a |

## PRIMARY CONTAINMENT LEAKAGE

### LIMITING CONDITION FOR OPERATION

3.6.1.2 DELETED

3/4 6-4 Deleted

SURVEILLANCE REQUIREMENTS (Continued)

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

1. CHANNEL CHECK at least once per 24 hours,

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

3. CHANNEL CALIBRATION at least once per 18 months.

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

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

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

c) High temperature $\leq$ 105°F

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

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

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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

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

4.6.3.6 At least once per 18 months:

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

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

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

DEPRESSURIZATION SYSTEMS (Continued)

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

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

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

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

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

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

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

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

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

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

The Technical Specifications contained in Appendix A, as revised through Amendment No. 87 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

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

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

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

(d) A one-time exemption from the requirement of paragraph III.A.6(b) of Appendix J to resume a Type A test schedule of three times in ten years. Exemptions (c) and (d) are described in the Safety Evaluation accompanying Amendment No. 87 to this license. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted pursuant to 10 CFR 50.12. With the granting of these exemptions the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

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. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted pursuant to 10 CFR 50.12. With the granting of these exemptions the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

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

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

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

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

ATTACHMENT TO LICENSE AMENDMENT NO. 87

FACILITY OPERATING LICENSE NO. NPF-18

DOCKET NO. 50-374

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

| REMOVE | INSERT |
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| I | I |
| II | II |
| VII | VII |
| XXII | XXII |
| 1-3 | 1-3 |
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| 1-5a | 1-5a |
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| 3/4 6-26 | 3/4 6-26 |
| 3/4 6-27 | 3/4 6-27 |
| 3/4 6-28 thru 6-37a | |
| 3/4 6-39 | 3/4 6-39 |
| *3/4 6-45 | *3/4 6-45 |
| 3/4 6-46 | 3/4 6-46 |
| B 3/4 6-1 | B 3/4 6-1 |
| B 3/4 6-2 | B 3/4 6-2 |
| | B 3/4 6-2a |
| B 3/4 6-4 | B 3/4 6-4 |
| | B 3/4 6-4a |

FRACTION OF LIMITING POWER DENSITY

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

FRACTION OF RATED THERMAL POWER

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

FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT SYSTEM

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

IDENTIFIED LEAKAGE

1.18 IDENTIFIED LEAKAGE shall be:

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

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

ISOLATION SYSTEM RESPONSE TIME

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

$L_a$

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

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

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

PRIMARY CONTAINMENT LEAKAGE

3.6.1.2 Deleted.

INTENTIONALLY LEFT BLANK

3/4 6-4 Deleted

SURVEILLANCE REQUIREMENTS (Continued)

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

1. CHANNEL CHECK at least once per 24 hours,

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

3. CHANNEL CALIBRATION at least once per 18 months.

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

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

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

c) High temperature $\leq$105°F

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

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

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

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

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

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

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

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

DEPRESSURIZATION SYSTEMS (Continued)

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

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

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

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

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

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

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