



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. 100  
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 20, 1993 and as supplemented by letters dated December 27, 1993, March 22, 1994, and May 31, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;
  - B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;
  - C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;
  - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
  - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-11 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 100, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

*Robert A. Capra*

Robert A. Capra, Director  
Project Directorate III-2  
Division of Reactor Projects - III/IV  
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical  
Specifications

Date of Issuance: June 3, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 100

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.

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- e. The suppression chamber is OPERABLE pursuant to Specification 3.6.2.1.
- f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.
- g. Primary containment structural integrity has been verified in accordance with Surveillance Requirement 4.6.1.1.e.

### PROCESS CONTROL PROGRAM

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

### PURGE - PURGING

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

### RATED THERMAL POWER

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

### REACTOR PROTECTION SYSTEM RESPONSE TIME

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

### REPORTABLE EVENT

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

### ROD DENSITY

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



### 3/4.6 CONTAINMENT SYSTEMS

#### 3/4.6.1 PRIMARY CONTAINMENT

##### PRIMARY CONTAINMENT INTEGRITY

##### LIMITING CONDITION FOR OPERATION

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3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

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

##### ACTION:

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

##### SURVEILLANCE REQUIREMENTS

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4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

- a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seal with gas at Pa, 39.6 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 La.
- b. At least once per 31 days by verifying that all primary containment penetrations\*\* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Table 3.6.3-1 of Specification 3.6.3.
- c. By verifying each primary containment air lock OPERABLE per Specification 3.6.1.3.
- d. By verifying the suppression chamber OPERABLE per Specification 3.6.2.1.
- e. Verify primary containment structural integrity in accordance with the Inservice Inspection Program for Post Tensioning Tendons. The frequency shall be in accordance with the Inservice Inspection Program for Post Tensioning Tendons.

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\*See Special Test Exception 3.10.1

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

3/4.6.1.5 INTENTIONALLY LEFT BLANK  
Pages 3/4 6-9 through 3/4 6-12 DELETED

## 3/4.6 CONTAINMENT SYSTEMS

### BASES

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#### 3/4.6.1 PRIMARY CONTAINMENT

##### 3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

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

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

##### 3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 39.6 psig,  $P_a$ . As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to  $0.75 L_a$  during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J to 10 CFR 50 with the exception of exemption(s) granted for main steam isolation valve leak testing and testing the airlocks after each opening.

##### 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 Specifications 3.6.1.1 and 3.6.1.2. 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.

## CONTAINMENT SYSTEMS

### BASES

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#### 3/4.6.1 PRIMARY CONTAINMENT

##### 3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR 100 guidelines provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIV's such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the isolation valves when isolation of the primary system and containment is required.

##### 3/4.6.1.5 DELETED

##### 3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

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

Onsite Review and Investigative Function (Continued)

b. Responsibility

The Onsite Review and Investigative Function shall be responsible for conducting the following:

- 1) Review of all applicable plant Administrative Procedures recommended in Appendix A of Reg Guide 1.33, Revision 2, February 1978 and changes thereto;
- 2) Review of Emergency Operating Procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No. 82-33 and changes thereto;
- 3) Review of all proposed tests and experiments that affect nuclear safety;
- 4) Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety;
- 5) Review of proposed changes to the Fire Protection Program;
- 6) Review of the Station Security Plan and submittal of recommended changes to the station Security Plan in accordance with station procedures;
- 7) Review of Emergency Plan and identification of recommended changes;
- 8) Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL;
- 9) Review of all proposed changes to the Technical Specifications or Operating License, and any proposed change which involves an unreviewed safety question that is to be submitted to the Commission for approval;
- 10) Review of investigation results for all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluations and recommendation to prevent recurrence;
- 11) Review of investigation results for all REPORTABLE EVENTS and other significant operating abnormalities including the preparation and forwarding of reports covering evaluations and recommendation to prevent recurrence.
- 12) Review of investigation results for any accidental, unplanned, or uncontrolled radioactive release including the preparation and forwarding of reports covering evaluations and recommendations to prevent recurrence;

Onsite Review and Investigative Function (Continued)

- 13) Review of Unit operations to detect potential hazards to nuclear safety;
- 14) Performance of special reviews and investigations and reports thereon as requested by the Superintendent of the Offsite Review and Investigative Function;
- 15) Review of changes to the Inservice Inspection Program for Post Tensioning Tendons.

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

- i. Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,
- j. Limitations on venting and purging of the containment through the Primary Containment Vent and Purge System or Standby Gas Treatment System to maintain releases as low as reasonably achievable,
- k. Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

5. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

- a. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,
- b. A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and
- c. Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

6. Inservice Inspection Program for Post Tensioning Tendons

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, Revision 3, 1989, except that the unit 1 and 2 primary containments shall be treated as twin containments even though the Initial Structural Integrity Tests were not within 2 years of each other.

The provisions of 4.0.2 and 4.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

6.3 ACTION TO BE TAKEN IN THE EVENT OF A REPORTABLE EVENT IN PLANT OPERATION

The following actions shall be taken for REPORTABLE EVENTS:

- a. The Commission shall be notified and a Licensee Event Report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and
- b. Each REPORTABLE EVENT shall be reviewed pursuant to Specification 6.1.G.2.c(1).



PLANT OPERATING RECORDS (Continued)

5. Records of plant radiation and contamination surveys;
6. Records of offsite environmental monitoring surveys;
7. Records of radiation exposure for all plant personnel, including all contractors and visitors to the plant, in accordance with 10 CFR Part 20;
8. Records of radioactivity in liquid and gaseous wastes released to the environment;
9. Records of transient or operational cycling for those components that have been designed to operate safely for a limited number of transient or operational cycles (identified in Table 5.7.1-1);
10. Records of individual staff members indicating qualifications, experience, training, and retraining;
11. Inservice inspections of the reactor coolant system;
12. Minutes of meetings and results of reviews and audits performed by the offsite and onsite review and audit functions;
13. Records of reactor tests and experiments;
14. Records of Quality Assurance activities required by the QA Manual, except for those items specified in Section 6.5.A;
15. Records of reviews performed for changes made to procedures on equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;
16. Records of the service lives of all hydraulic and mechanical snubbers required by specification 3.7.9 including the date at which the service life commences and associated installation and maintenance records;
17. Records of analyses required by the radiological environmental monitoring program;
18. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM; and
19. Records of pre-stressed concrete containment tendon surveillances.

6.6 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted



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. 84  
License No. NPF-18

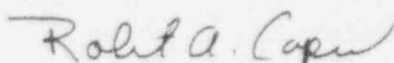
1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment filed by the Commonwealth Edison Company (the licensee), dated August 20, 1993 and as supplemented by letters dated December 27, 1993, March 22, 1994, and May 31, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;
  - B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;
  - C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;
  - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
  - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-18 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 84 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION



Robert A. Capra, Director  
Project Directorate III-2  
Division of Reactor Projects - III/IV  
Office of Nuclear Reactor Regulation

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: June 3, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 84

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.

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## DEFINITIONS

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### OPERATIONAL CONDITION - CONDITION

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

### PHYSICS TESTS

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

### PRESSURE BOUNDARY LEAKAGE

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

### PRIMARY CONTAINMENT INTEGRITY

1.31 PRIMARY CONTAINMENT INTEGRITY shall exist when:

- a. All primary containment penetrations required to be closed during accident conditions are either:
  1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or
  2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 of Specification, 3.6.3.
- b. All primary containment equipment hatches are closed and sealed.
- c. Each primary containment air lock is OPERABLE pursuant to Specification 3.6.1.3.
- d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.
- e. The suppression chamber is OPERABLE pursuant to Specification 3.6.2.1.
- f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.
- g. Primary containment structural integrity has been verified in accordance with Surveillance Requirement 4.6.1.1.e.



## DEFINITIONS

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### PROCESS CONTROL PROGRAM

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

### PURGE - PURGING

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

### 3/4.6 CONTAINMENT SYSTEMS

#### 3/4.6.1 PRIMARY CONTAINMENT

##### PRIMARY CONTAINMENT INTEGRITY

##### LIMITING CONDITION FOR OPERATION

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3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

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

##### SURVEILLANCE REQUIREMENTS

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4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

- a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seal with gas at Pa, 39.6 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 La.
- b. At least once per 31 days by verifying that all primary containment penetrations\*\* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Table 3.6.3-1 of Specification 3.6.3.
- c. By verifying each primary containment air lock OPERABLE per Specification 3.6.1.3.
- d. By verifying the suppression chamber OPERABLE per Specification 3.6.2.1.
- e. Verify primary containment structural integrity in accordance with the Inservice Inspection Program for Post Tensioning Tendons. The frequency shall be in accordance with the Inservice Inspection Program for Post Tensioning Tendons.

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\*See Special Test Exception 3.10.1

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

3/4.6.1.5 INTENTIONALLY LEFT BLANK  
Pages 3/4 6-9 through 3/4 6-15 DELETED

## 3/4.6 CONTAINMENT SYSTEMS

### BASES

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#### 3/4.6.1 PRIMARY CONTAINMENT

##### 3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

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

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

##### 3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 39.6 psig,  $P_a$ . As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to  $0.75 L_a$  during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J to 10 CFR 50 with the exception of exemption(s) granted for main steam isolation valve leak testing and testing the airlocks after each opening.

##### 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 Specifications 3.6.1.1 and 3.6.1.2. 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.

## CONTAINMENT SYSTEMS

### BASES

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#### 3/4.6.1 PRIMARY CONTAINMENT

##### 3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR 100 guidelines provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIV's such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the isolation valves when isolation of the primary system and containment is required.

##### 3/4.6.1.5 DELETED

##### 3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

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

- 12) Review of investigation results for any accidental, unplanned, or uncontrolled radioactive release including the preparation and forwarding of reports covering evaluations and recommendations to prevent recurrence;
- 13) Review of Unit operations to detect potential hazards to nuclear safety;
- 14) Performance of special reviews and investigations and reports thereon as requested by the Superintendent of the Offsite Review and Investigative Function;
- 15) Review of changes to the Inservice Inspection Program for Post Tensioning Tendons.

c. Authority

The Onsite Review and Investigative Function shall:

- 1) Advise the Station Manager on all matters related to Nuclear Safety;
- 2) Recommend to the Station Manager the disposition of items considered under Specification 6.1.G.2.b.1) through 9) prior to their implementation;
- 3) Include among its review conclusions for each item considered under Specification 6.1.G.2.b.1) through 4), a determination of whether or not the item involves an unreviewed safety question.
- 4) Provide prompt notification to the Vice-President BWR Operations and the Superintendent of the Offsite Review and Investigative Function of disagreement between the Onsite Review and Investigative Function and the Station Manager. The Station Manager shall follow the recommendations of the Onsite Review and Investigative Function or select a course of action that is more conservative regarding safe operation of the facility.

d. Records

- 1) Reports, reviews, investigations, and recommendations prepared and performed for Specification 6.1.G.2a shall be documented and forwarded to the Superintendent of the Offsite Review and Investigative Function unless otherwise specified.
- 2) Copies of all records and documentation shall be kept on file at the station.

e. Procedures

Written administrative procedures shall be prepared and maintained for conduct of the Onsite Review and Investigative Function. These procedures shall include the following:

- 1) Content and method of submission and presentation to the Station Manager, Vice President BWR Operations, and the Superintendent of the Offsite Review and Investigative Function.

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

- i. Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,
- j. Limitations on venting and purging of the containment through the Primary Containment Vent and Purge System or Standby Gas Treatment System to maintain releases as low as reasonably achievable,
- k. Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

5. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

- a. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,
- b. A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and
- c. Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

6. Inservice Inspection Program for Post Tensioning Tendons

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, Revision 3, 1989, except that the unit 1 and 2 primary containments shall be treated as twin containments even though the Initial Structural Integrity Tests were not within 2 years of each other.

The provisions of 4.0.2 and 4.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

6.3 ACTION TO BE TAKEN IN THE EVENT OF A REPORTABLE EVENT IN PLANT OPERATION

The following actions shall be taken for REPORTABLE EVENTS:

- a. The Commission shall be notified and a Licensee Event Report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and
- b. Each REPORTABLE EVENT shall be reviewed pursuant to Specification 6.1.G.2.c(1).



PLANT OPERATING RECORDS (Continued)

9. Records of transient or operational cycling for those components that have been designed to operate safely for a limited number of transient or operational cycles (identified in Table 5.7.1-1);
10. Records of individual staff members indicating qualifications, experience, training, and retraining;
11. Inservice inspections of the reactor coolant system;
12. Minutes of meetings and results of reviews and audits performed by the offsite and onsite review and audit functions;
13. Records of reactor tests and experiments;
14. Records of Quality Assurance activities required by the QA Manual, except for those items specified in Section 6.5.A;
15. Records of reviews performed for changes made to procedures on equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;
16. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.9 including the date at which the service life commences and associated installation and maintenance records;
17. Records of analyses required by the radiological environmental monitoring program;
18. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM; and
19. Records of pre-stressed concrete containment tendon surveillances.

6.6 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the director of the appropriate Regional Office of Inspection and Enforcement unless otherwise noted.

A. Routine Reports

1. Startup Report

A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.