Docket File

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

June 3, 1994

Docket Nos. 50-373
and 50-374

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

Dear Mr. Farrar:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M87305 AND M87306)

The Commission has issued the enclosed Amendment No. 100 to Facility Operating License No. NPF-11 and Amendment No. 84 to Facility Operating License No. NPF-18 for the LaSalle County Station, Units 1 and 2, respectively. The amendments are in response to your application dated August 20, 1993 and as supplemented by letters dated December 27, 1993, March 22, 1994, and May 31, 1994.

The amendments delete Technical Specification Section 3/4.6.1.5, "Primary Containment Structural Integrity" which includes Surveillance Requirements for the Primary Containment Tendons and adds a Technical Specification requirement to establish, implement, and maintain a comprehensive containment tendon program. The containment tendon program is based on Regulatory Guide 1.35, Rev.3 and is titled, "Inservice Inspection Program for Post Tensioning Tendons." The new program will allow the Unit 1 and 2 containments to be tested as one containment.

As discussed with Messrs. Lockwood and Benes of your staff on June 2, 1994, the staff is concerned that your March 22, 1994 submittal did not highlight and thoroughly discuss new design bases information. Furthermore, you did not indicate explicitly why it was not necessary to revise the no significant hazards consideration. This calls into question the adequacy of your on-site review process. Further details are provided in the introduction section of the attached Safety Evaluation.

DF01
1/1

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

Sincerely,

Original Signed By:

Anthony T. Gody, Jr., Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 100 to NPF-11
2. Amendment No. 84 to NPF-18
3. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION:

Docket File	NRC & Local PDRs
J. Roe	J. Zwolinski
R. Capra	C. Hawes
A. Gody, Jr.	OGC
D. Hagan	PDIII-2 r/f
C. Grimes	ACRS (10)
OPA	OC/LFDCB
G. Hill (4)	B. Clayton RIII
C. Grimes	G. Bagchi

OFC	LA:PDIII-2	PM:PDIII-2	BC:ECGB	BC:OTSB	D:PDIII-2	OGC
NAME	CHAWES CMH	AGODY AT	GBAGCHI*	CGRIMES*#	RCAPRA Rac	*
DATE	6/2/94	6/3/94	05/18/94	05/18/94	6/3/94	05/19/94
COPY	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO

*See previous concurrence # Concur via E-mail

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Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III, Suite 500
1400 OPUS Place
Downers Grove, Illinois 60515

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SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M87305 AND M87306)

The Commission has issued the enclosed Amendment No. to Facility Operating License No. NPF-11 and Amendment No. to Facility Operating License No. NPF-18 for the LaSalle County Station, Units 1 and 2, respectively. The amendments are in response to your application dated August 20, 1993 and as supplemented by letters dated December 27, 1993 and March 22, 1994.

The amendments delete Technical Specification Section 3/4.6.1.5, "Primary Containment Structural Integrity which include Surveillance Requirements for the Primary Containment Tendons and adds a Technical Specification requirement to establish, implement, and maintain a comprehensive containment tendon program. The containment tendon program is based on Regulatory Guide (RG) 1.35, Revision (Rev.) 3, and is titled "Inservice Inspection Program for Post Tensioning Tendons." The new program will allow Units 1 and 2 containments to be tested as twin containments.

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

Sincerely,

Anthony T. Gody, Jr., Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. to NPF-11
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cc w/enclosures:
See next page

Subject to noted w/changes

OFC	LA:PDIII-2	PM:PDIII-2	BC:ECGB	BC:OTSB	D:PDIII-2	OGC
NAME	CHAWES CMH	AGODY AG	GBAGCHI for	CGRIMES*	~~JDYER~~ RCapra	
DATE	5/18/94	5/18/94	5/18/94	5/18/94	/ /94	5/19/94
COPY	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO

*Concur via E-mail on 5/18/94 AT

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

Sincerely,

Anthony T. Gody, Jr.

Anthony T. Gody, Jr., Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

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NAME	CHAWES CMH	AGODY AT	GBAGCHI*	CGRIMES*#	RCAPRA Ra	*
DATE	6/2/94	6/3/94	05/18/94	05/18/94	6/3/94	05/19/94
COPY	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO

*See previous concurrence # Concur via E-mail

Mr. D. L. Farrar
Commonwealth Edison Company

LaSalle County Station
Unit Nos. 1 and 2

cc:

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-373

LASALLE COUNTY STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 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.

REMOVE	INSERT
VII	VII
XIV	XIV
XXII	XXII
XXIII	XXIII
1-6	1-6
3/4 6-1	3/4 6-1
3/4 6-8	3/4 6-8
3/4 6-9	-
3/4 6-10	-
3/4 6-11	-
3/4 6-12	-
3/4 6-12a	-
3/4 6-12b	-
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2
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-	6-9a
6-20	6-20
-	6-20a
6-22	6-22

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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BASES

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3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.6 CONTAINMENT SYSTEMS

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

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.

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. 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.

*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

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.

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 safety 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

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.

REMOVE	INSERT
VII	VII
XIV	XIV
XXII	XXII
1-5	1-5
-	1-5a
3/4 6-1	3/4 6-1
3/4 6-8	3/4 6-8
3/4 6-9	-
3/4 6-10	-
3/4 6-11	-
3/4 6-12	-
3/4 6-13	-
3/4 6-14	-
3/4 6-15	-
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2
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-	6-20a
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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

SECTION | PAGE

LIST OF TABLES (Continued)

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

PRESSURE BOUNDARY LEAKAGE

1.30 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.

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

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. 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.

*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

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.

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 radioative 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 safety 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 100 TO FACILITY OPERATING LICENSE NO. NPF-11 AND

AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NO. NPF-18

COMMONWEALTH EDISON COMPANY

LASALLE COUNTY STATION, UNITS 1 AND 2

DOCKET NOS. 50-373 AND 50-374

1.0 INTRODUCTION

By letter dated August 20, 1993, Commonwealth Edison Company (CECo, the licensee) requested that the Nuclear Regulatory Commission (NRC) approve a change to the LaSalle County Station (LSCS) Units 1 and 2 Technical Specification (TS) 3/4.6.1.5, "Primary Containment Structural Integrity." The amendment request (a) deletes TS 3/4.6.1.5 and relocates the containment surveillance requirement by adding TS Administrative Control 6.2.F.6, "Plant Operating Procedures and Programs," to require a containment tendon program titled, "Inservice Inspection Program for Post Tensioning Tendons," be established, implemented, and maintained; (b) modifies the TS Bases to include a brief description of the Inservice Inspection Program for Post Tensioning Tendons; (c) adds a requirement in TS 6.1.G.2.b, "Organization, Review, Investigation, and Audit," for onsite review of changes to the Inservice Inspection Program for Post Tensioning Tendons; (d) requires the licensee to maintain records of pre-stressed concrete containment tendon surveillance data in TS 6.5.B, "Plant Operating Records"; and (e) considers the two units as twin units by taking exception to certain provisions of Regulatory Guide (RG) 1.35, Revision (Rev.) 3.

Following the initial review of the licensee's submittal, on December 2, 1993, the staff issued a request for information concerning (a) data to support the technical justification for the exception to RG 1.35, Rev. 3 concerning Inservice Inspection (ISI) interval, and (b) the characterization of the need for amending the TSs. The licensee responded by letter dated December 27, 1993.

As the staff's review continued, additional questions concerning the original submittal and the supplemental letter were identified and raised to the licensee in a public meeting on February 15, 1994. These questions involved (a) a plot of all the tendon lift-off data in graphical form similar to that described in RG 1.35, Rev. 3; (b) a clarification of predicted lift-off force differences between the two units since part of the amendment involved justification of containment similarity; (c) a clarification of how the Limiting Condition for Operation for TS 3.6.1.1 will be applied to new TS Surveillance Requirement 4.6.1.1.e; (d) clarification of the applicability of

Appendix B of the original submittal; (e) a request to see the planned updated Final Safety Analysis Report section concerning the Inservice Inspection Program for Post Tensioning Tendons which includes appropriate acceptance criteria; (f) a request to include the requirement for primary containment structural integrity in the TS definition of PRIMARY CONTAINMENT INTEGRITY, and (g) a discussion of the prestress loss considered in the design of the containment structure, particularly the time-dependent prestress loss due relaxation, creep, and shrinkage. Most of these questions were resolved directly in the public meeting, however, several were deferred for further review by the licensee. By letter dated March 22, 1994, the licensee provided answers to the deferred questions from the February 15, 1994 public meeting.

In a phone conversation on May 16, 1994, the licensee indicated that they would include in their UFSAR update, a description of how they would trend the tendon forces for each group of tendons as obtained through tendon lift-off surveillance. The licensee indicated that they would, utilizing linear regression analysis after each lift-off surveillance, plot on one graph, all existing and new lift-off force data for each group for the LaSalle twin containments. The date of completion for each unit will be the common starting point and the data plotted will be based on the number of years and months from this same starting point. During this phone conversation and another on May 19, 1994, the staff had questions concerning the design bases reflected in the licensee's March 22, 1994 submittal for measured tendon forces (575 kips hoop and 600 kips vertical) which were different from those described in the February 15, 1994 public meeting (620 kips hoop and 626 kips vertical). Item "c" of Attachment A to the licensee's March 22, 1994 submittal (Item C) indicates that "effective tendon end anchor forces at the end of 40 years are 620 kips... in the hoop tendons and 626 kips... in the vertical tendons", and that "[t]hese values are above the minimum required 40-year tendon end anchor forces of 575 kips in the hoop tendons and 600 kips in the vertical tendons." Item C indicates that the 40-year design bases for hoop and vertical tendon forces differ from "effective" tendon forces. However, more importantly, the staff recognized during phone conversations on May 16, and May 19, 1994 that these new design bases were a result of a new design calculation dated March 14, 1994. The staff was concerned that Item C does not indicate that these "40-year tendon end anchor force" design bases were developed in a LaSalle Station design calculation dated March 14, 1994 and the "effective tendon end anchor forces at the end of 40 years" were actually the old design bases. Further, the licensee's submittal did not address the impact of these new design bases on the original no significant hazards consideration or indicate whether this design change had, or was planned to be implemented in accordance with the provisions of 10 CFR § 50.59.

On May 31, 1994, the licensee provided a written response to the above mentioned concerns which were identified in the staff's review of the licensees March 22, 1994 letter. The May 31 letter clearly indicates that the new design bases information was not intended to be part of the license amendment request. The licensee indicated the proposed Inservice Inspection Program for Post Tensioning Tendons and any associated UFSAR change are not currently being used and will not be able to be used until the proposed

Technical Specification amendment is approved by the NRC. Any changes to the program and/or the UFSAR will be made in accordance with 10 CFR § 50.59.

2.0 BACKGROUND

10 CFR § 50.36(c) requires that TS include items in five specific categories, including safety limits, limiting safety system settings, limiting control settings; limiting conditions for operation; surveillance requirements; design features; and administrative controls. However, the rule does not specify the particular requirements to be included in a plant's TS. The NRC has developed guidance criteria, as described in the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," 58 FR 39132 (July 22, 1993), which can be used to determine which of the design conditions and associated surveillance provisions need to be located in the TS. As stated therein, the TS must include those conditions or limitations on reactor operation which are "necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety." Briefly stated, those criteria are (1) detection of abnormal degradation of the reactor coolant pressure boundary; (2) conditions for bounding design basis accident and transient analyses; (3) primary success paths to prevent or mitigate design basis accidents and transients; and (4) functions determined to be important to risk or operating experience. The policy statement recognized that items which are relocated from the TS to licensee-controlled documents such as the updated FSAR would in turn be controlled in accordance with the requirements of 10 CFR § 50.59, "Changes, tests, and experiments." 10 CFR § 50.59 provides criteria to determine when facility or operating changes planned by a licensee require prior Commission approval in the form of a license amendment in order to address any unreviewed safety questions. NRC inspection and enforcement programs also enable the staff to monitor facility changes and licensee adherence to FSAR commitments and to take any remedial action that may be appropriate.

3.0 EVALUATION

The licensee's proposed deletion of TS 3/4.6.1.5 and addition of the requirements of proposed TS 4.6.1.1.e, relocation of the containment tendon testing requirements to TS 4.6.1.1.e and removal of the containment tendon testing program from the TS and placement in licensee-controlled documents, in this case, the UFSAR and the Inservice Inspection Program for Post Tensioning Tendons. These licensee-controlled documents would, in turn, be controlled in accordance with the requirements of 10 CFR § 50.59. To assure that changes to the Inservice Inspection Program for Post Tensioning Tendons are appropriately reviewed, the licensee also proposed a requirement in TS 6.1.G.2.b for onsite review of changes to the Inservice Inspection Program for Post Tensioning Tendons. Record keeping requirements have been proposed in TS 6.5.B.

The proposed revision to the LSCS UFSAR includes (a) a description of the tendon surveillance testing history and conversion to the new test frequencies; (b) a description summary of the Inservice Inspection Program for Post Tensioning Tendons, and (c) acceptance criteria.

The licensee's summary of the Inservice Inspection Program for Post Tensioning Tendons indicates that the program specifies procedures, methodologies, and acceptance criteria for (a) visual inspection; (b) tendon anchorage areas inspection; (c) prestress monitoring tests; (d) tendon material test and inspection; and (e) filler grease inspection, to ensure that potential degradation of the primary containment is detected and appropriate corrective actions are taken.

The staff has reviewed the acceptance criteria contained within the UFSAR and concludes that they meet the provisions of RG 1.35, Rev. 3. Experience has shown that structural degradation of the primary containment is a predictable process that can be monitored by a comprehensive containment tendon monitoring program. On this basis, the staff concludes that the Inservice Inspection Program for Post Tensioning Tendons is not required to be in the TS by 10 CFR § 50.36 or to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety, and that changes to the design conditions and surveillance provisions can be made by the licensee consistent with the procedures and controls imposed by 10 CFR § 50.59.

In addition to requesting a license amendment that would relocate the containment tendon testing program from the TS to the UFSAR and Inservice Inspection Program for Post Tensioning Tendons, the licensee has requested that the following change concerning twin containments be made:

> Regulatory Guide 1.35, Rev. 3, has provisions for In-Service Inspection (ISI) frequency of 1, 3, 5 and every 5 years thereafter for twin unit containments. This provision allows the units to be tested alternately, so that tests are performed (except for visual inspections and grease samples) every 10 years for each unit with approximately 5 years between the tests for the units combined. One of the conditions listed in Regulatory Guide 1.35, Rev. 3, for treating the dual unit primary containments as twin containments states:
>
> > "1.5.b. Their [the containments'] ISITs [Initial Structural Integrity Tests] were performed within two years of each other."
>
> The licensee seeks an exception to this provision in R.G. 1.35, Rev. 3.

The ISITs for the LaSalle Unit 1 and Unit 2 Primary Containments were approximately 4.5 years apart. The licensee indicated that complete ISIs of Unit 1 and Unit 2 performed to date demonstrate that this 2 year limit on ISITs is not a factor influencing LaSalle Unit 1 and Unit 2 tendon integrity.

The staff requested the licensee to provide information on the behavior of the containments during the SIT of each unit and on the lift-off forces of the tendons selected for the ISI tests, six in number for Unit 1 and four for Unit 2. The staff reviewed the data and concludes that there is reasonable agreement in the deflection values obtained during SITS at comparable locations of the containments. For the lift-off forces, if using the completion of construction dates as starting points for comparison, the differences between two units are of little significance. This should be expected since the difference in the completion dates is only two years. On the basis of this observation, the staff concurs with the licensee that the two units can be considered as twin units and approves the licensee's application for the amendment to consider the two units as twin units, as an exception to R.G. 1.35, Rev. 3, 1.5.b.

The licensee stated that a limiting condition for operation (LCO) will be entered for each unit if primary containment is determined to be inoperable on the basis of the results of one unit's tendon surveillance.

The staff has concluded that the licensee's proposed TS changes to LSCS Units 1 and 2 will provide adequate control of the Inservice Inspection Program for Post Tensioning Tendons. Further, the staff concludes after reviewing the tendon test data provided by the licensee that all of the changes regarding the interval for performing testing and treating the LaSalle Station containments as "twin containments" is acceptable.

LSCS containment tendon testing will be controlled by the Inservice Inspection Program for Post Tensioning Tendons, which is described in the licensee's UFSAR. In its letter dated March 22, 1994, the licensee committed to include these changes in the next annual update of its UFSAR. Any changes to the UFSAR or Inservice Inspection Program for Post Tensioning Tendons will be evaluated by the Onsite Review and Investigative Function and, if the changes are determined to involve an unreviewed safety question, the licensee will be required to submit a license amendment application to obtain prior NRC review and approval in accordance with 10 CFR § 50.59. The staff has concluded, therefore, that control of these tests in the TS (a) is not specifically required by 10 CFR § 50.36 or other regulations; (b) is not required to avert an immediate threat to the public health and safety; and (c) is not necessary because changes that are deemed to involve an unreviewed safety question will still require prior NRC approval by license amendment as provided by 10 CFR 50.59(c).

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (58 FR 59746). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

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