DOCKET FILE

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

July 7, 1989

Docket Nos. 50-373
and 50-374

Mr. Thomas J. Kovach
Nuclear Licensing Manager
Commonwealth Edison Company
P.O. Box 767
Chicago, Illinois 60690

Dear Mr. Kovach:

Subject: ISSUANCE OF AMENDMENTS NOS. 67 AND 49 TO FACILITY OPERATING LICENSE NOS. NPF-11 AND NPF-18 - LASALLE COUNTY STATION, UNITS 1 AND 2 (TAC NOS. 69067 AND 69068)

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 67 to Facility Operating License No. NPF-11 and Amendment No. 49 to Facility Operating License No. NPF-18 for the LaSalle County Station, Units 1 and 2. These amendments are in response to your letter dated October 7, 1988.

The amendments revise the LaSalle County Station, Units 1 and 2 Technical Specifications by allowing operation of both units with suppression pool temperatures of up to 105°F. The current suppression pool temperature limit during normal operation is 100°F.

A copy of the related Safety Evaluation supporting Amendment No. 67 to Facility Operating License No. NPF-11 and Amendment No. 49 to Facility Operating License No. NPF-18 is enclosed. The Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Paul C. Shemanski

Paul C. Shemanski, Project Manager
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:
1. Amendment No. 67 to NPF-11
2. Amendment No. 49 to NPF-18
3. Safety Evaluation

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cc w/enclosure:
See next page

July 7, 1989

Docket Nos. 50-373
and 50-374

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Mr. Thomas J. Kovach
Nuclear Licensing Manager
Commonwealth Edison Company
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Dear Mr. Kovach:

Subject: ISSUANCE OF AMENDMENTS NOS. 67 AND 49 TO FACILITY OPERATING LICENSE NOS. NPF-11 AND NPF-18 - LASALLE COUNTY STATION, UNITS 1 AND 2 (TAC NOS. 69067 AND 69068)

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IV, V and Special Projects

Enclosures:
1. Amendment No. 67 to NPF-11
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3. Safety Evaluation

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See next page

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SUBJECT TO CHANGES

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-373

LASALLE COUNTY STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 67
License No. NPF-11

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the Commonwealth Edison Company (the licensee), dated October 7, 1988 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. 67 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Paul C. Shemanski, Acting Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: July 7, 1989

ENCLOSURE TO LICENSE AMENDMENT NO. 67

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
3/4 6-16	3/4 6-16
3/4 6-17	3/4 6-17
3/4 6-18	3/4 6-18

SUPPRESSION CHAMBER#

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. Volume between 131,900 ft^3 and 128,800 ft^3, equivalent to a level between +3 inches** and -4 1/2 inches**, and a

2. Maximum average temperature of 105°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

a) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

b) 120°F with the main steam line isolation valves closed following a scram.

b. Drywell-to-suppression charmer bypass leakage less than or equal to 10% of the acceptable $A/\sqrt{k}$ design value of 0.03 ft^2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 1 or 2 with the suppression chamber average water temperature greater than or equal to 105°F, stop all testing which adds heat to the suppression pool, and restore the average temperature to less than or equal to 105°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression chamber average water temperature greater than 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

2. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

#See Specification 3.5.3 for ECCS requirements.
**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With one suppression chamber water level instrumentation channel inoperable and/or with one suppression pool water temperature instrumentation division inoperable, restore the inoperable instrumentation to OPERABLE status within 7 days or verify suppression chamber water level and/or temperature to be within the limits at least once per 12 hours by local indication.

d. With both suppression chamber water level instrumentation channels inoperable and/or with both suppression pool water temperature instrumentation divisions inoperable, restore at least one inoperable water level channel and one water temperature division to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With the drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

a. By verifying the suppression chamber water volume to be within the limits at least once per 24 hours.

b. At least once per 24 hours in OPERATIONAL CONDITION 1 or 2 by verifying the suppression chamber average water temperature to be less than or equal to 105°F, except:

1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature less than or equal to 105°F.

2. At least once per 60 minutes when suppression chamber average water temperature is greater than 105°F, by verifying suppression chamber average water temperature less than or equal to 110°F and THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

3. At least once per 30 minutes following a scram with suppression chamber average water temperature greater than or equal to 105°F, by verifying suppression chamber average water temperature less than or equal to 120°F.

SURVEILLANCE REQUIREMENTS (Continued)

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

1. CHANNEL CHECK at least once per 24 hours,

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

3. CHANNEL CALIBRATION at least once per 18 months.

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

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

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

c) High temperature $\leq$ 105°F

d. By conducting drywell-to-suppression chamber bypass leak tests and verifying that the $A/\sqrt{k}$ calculated from the measured leakage is within the specified limit when drywell-to-suppression chamber bypass leak tests are conducted:

1. At least once per 18 months at an initial differential pressure of 1.5 psi, and

2. At the first refueling outage and then on the schedule required for Type A Overall Integrated Containment Leakage Rate tests by Specification 4.6.1.2.a; at an initial differential pressure of 5 psi,

except that, if the first two 5 psi leak tests performed up to that time result in:

1. A calculated $A/\sqrt{k}$ within the specified limit, and

2. The $A/\sqrt{k}$ calculated from the leak tests at 1.5 psi is $\leq$ 20% of the specified limit,

then the leak tests at 5 psi may be discontinued.

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

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-374

LASALLE COUNTY STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 49
License No. NPF-18

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the Commonwealth Edison Company (the licensee), dated October 7, 1988 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. 49 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Paul C. Shemanski, Acting Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: July 7, 1989

ENCLOSURE TO LICENSE AMENDMENT NO. 49

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
3/4 6-19	3/4 6-19
3/4 6-20	3/4 6-20
3/4 6-21	3/4 6-21

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER#

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. Volume between 131,900 ft^3 and 128,800 ft^3, equivalent to a level between +3 inches** and -4 1/2 inches**, and a

2. Maximum average temperature of 105°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

a) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

b) 120°F with the main steam line isolation valves closed following a scram.

b. Drywell-to-suppression chamber bypass leakage less than or equal to 10% of the acceptable $A/\sqrt{k}$ design value of 0.03 ft^2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 1 or 2 with the suppression chamber average water temperature greater than or equal to 105°F, stop all testing which adds heat to the suppression pool, and restore the average temperature to less than or equal to 105°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression chamber average water temperature greater than 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

2. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

#See Specification 3.5.3 for ECCS requirements.
**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With one suppression chamber water level instrumentation channel inoperable and/or with one suppression pool water temperature instrumentation division inoperable, restore the inoperable instrumentation to OPERABLE status within 7 days or verify suppression chamber water level and/or temperature to be within the limits at least once per 12 hours by local indication.

d. With both suppression chamber water level instrumentation channels inoperable and/or with both suppression pool water temperature instrumentation divisions inoperable, restore at least one inoperable water level channel and one water temperature division to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With the drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

a. By verifying the suppression chamber water volume to be within the limits at least once per 24 hours.

b. At least once per 24 hours in OPERATIONAL CONDITION 1 or 2 by verifying the suppression chamber average water temperature to be less than or equal to 105°F, except:

1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature less than or equal to 105°F.

2. At least once per 60 minutes when suppression chamber average water temperature is greater than 105°F, by verifying suppression chamber average water temperature less than or equal to 110°F and THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

3. At least once per 30 minutes following a scram with suppression chamber average water temperature greater than or equal to 105°F, by verifying suppression chamber average water temperature less than or equal to 120°F.

SURVEILLANCE REQUIREMENTS (Continued)

c. By verifying at least 2 suppression chamber water level instrumentation channels and at least 14 suppression pool water temperature instrumentation channels, 7 in each of two divisions, OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,

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

3. CHANNEL CALIBRATION at least once per 18 months.

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

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

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

c) High temperature $\leq$ 105°F

d. By conducting drywell-to-suppression chamber bypass leak tests and verifying that the $A/\sqrt{k}$ calculated from the measured leakage is within the specified limit when drywell-to-suppression chamber bypass leak tests are conducted:

1. At least once per 18 months at an initial differential pressure of 1.5 psi, and

2. At the first refueling outage and then on the schedule required for Type A Overall Integrated Containment Leakage Rate tests by Specification 4.6.1.2.a., at an initial differential pressure of 5 psi,

except that, if the first two 5 psi leak tests performed up to that time result in:

1. A calculated $A/\sqrt{k}$ within the specified limit, and

2. The $A/\sqrt{k}$ calculated from the leak tests at 1.5 psi is $\leq$ 20% of the specified limit,

then the leak tests at 5 psi may be discontinued.

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 67 TO FACILITY OPERATING LICENSE NO. NPF-11 AND

AMENDMENT NO. 49 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 October 7, 1988, Commonwealth Edison Company, the licensee for LaSalle County Station, Units 1 and 2 requested changes to Technical Specifications (TS) 3.4.6.2.1 and 4.6.2.1. These specifications deal with the limiting conditions of operation (LCO) of the suppression pool (SP) during normal plant operation at conditions 1, 2 and 3 and the associated surveillance requirement for both the units. Specifically, the proposed change would raise the suppression pool temperature limit during normal operation from 100°F to 105°F. The 105°F limit on allowable pool temperature during safety system testing, which adds heat to the suppression pool, will not be changed. Also, the suppression pool temperature limit (SPTL), requiring immediate plant shutdown (110°F) and vessel depressurization (120°F), will remain unchanged.

The licensee stated that the unusually high temperatures in Illinois, the temperature of the LaSalle lake, which serves as the ultimate heat sink for the plant service water and residual heat removal (RHR) systems, have risen to the point where an insufficient differential temperature is available to maintain the suppression pool temperature below 100°F. In support of this increase in the suppression pool temperature limit during normal operation, the licensee provided the General Electric (GE) Company's Safety Evaluation (EAS-49-0888, Revision 1 dated August, 1988) of the suppression pool temperature limit for Mark II containment and its applicability to LaSalle Units. The GE report discussed the impact of the proposed increase in the pool's operational temperature limit on (1) containment response, (2) safety-relief valve (SRV) operation, (3) emergency core cooling system (ECCS) performance, (4) NPSH for safety system pumps, (5) LaSalle emergency operating procedures (EOPs), and (6) an anticipated transient without scram (ATWS) evaluations.

2.0 EVALUATION

The events which involve the suppression pool can be divided into two general categories: safety relief valve (SRV) discharge to the pool via the SRV discharge lines and T-Quenchers, and discharges to the pool via the drywell to wetwell vent pipes during design basis loss-of-coolant accidents (LOCA). The following discussion addresses the LOCA-related containment loads and the SRV operational loads.

2.1 LOCA-Related Containment Loads

The GE Safety Evaluation of the suppression pool temperature limit for LaSalle Units 1 and 2 discussed the ranges for operational temperature limits for SP water under LOCA conditions to ensure that containment pressures and temperatures and hydrodynamic loads under such conditions do not exceed the design values. The GE evaluation concludes that the normal operating suppression pool temperature up to 110°F for the LaSalle units will not affect its design load. The following paragraphs (a) through (d) summarize these evaluations and discuss their application to the LaSalle units.

(a) Containment Pressure and Temperature Design Limits

The GE report compared the pressure and temperature design limits for several Mark II plants (including LaSalle) to the predicted maximum containment pressure and temperatures during a LOCA. The report noted that because the design limits are very high for such containments, there is a large margin between the predicted values under LOCA conditions and the design values that would support a large increase in the normal operational pool temperature. Therefore, the report concluded that an increase in the operational pool temperature limit to 105°F will not impact the existing analytical results.

(b) Steam Condensation

With regard to the ability of the suppression pool to ensure complete steam condensation following a LOCA, the report stated that based on an analysis of test data for the Mark I (NEDE-24539P) and Mark II (NEDE-24811P) full scale test facility (FSTF), GE determined that a normal operational pool temperature in the range of 210°F to 220°F would ensure complete steam condensation because it would correspond to the tested maximum pool temperatures for which complete steam condensation was confirmed.

(c) Condensation Oscillation Loads

The report pointed out that condensation oscillation (CO) loads are primarily affected by two hydrodynamic parameters (NEDE-24288P), i.e., pool temperature and the enthalpy flux through the downcomer vents. Using the GE-developed correlation between these two parameters and the CO loads under transient conditions, the CO loads for the expected LOCA conditions and the conditions simulated during the FSTF test were determined and compared with plant-specific predictions to determine the margin between the expected and the design CO loads and, subsequently, the associated margin in the pool temperature. The licensee stated that consideration of LaSalle plant-specific bounding hydrodynamic parameters would result in a CO load that is less than that assumed in containment loads evaluation even with a normal operational pool temperature of 110°F.

(d) Chugging Loads

The GE report stated that a review of chugging data obtained during the Mark II FSTF tests (NEDE 24811-P) indicated that chugging occurs only with small-break LOCAs and relatively low pool temperatures (less than 135°F). The report concluded that the proposed increase in the normal operational pool temperature limit will have no impact on chugging loads.

On the basis of the GE information, the staff concludes that the LOCA-related containment loads resulting from the proposed increase in normal operational pool temperature limit will be within the containment design loads.

2.2 SRV Operational Loads

The SRV operational loads can be divided into two categories; the SRV air clearing load and the SRV condensation loads.

(a) SRV Air Clearing Loads

The SRV air clearing loads result from the expulsion of air out of the SRV discharge line into the suppression pool. These loads are defined in NUREG-0802, "SRV Quencher Loads Evaluation for BWR Mark II and Mark III Containments." The expansion and contraction of the air bubble creates an oscillatory load on the containment wall and submerged structures. The SRV air clearing load will increase with a higher initial pool temperature based on a review of test data (NEDE-21078P). However, the staff notes that the US Mark II containment program requires that the limiting SRV air clearing load to be considered in containment structural evaluations be determined on the basis of the first actuation of an SRV at the maximum pool temperature permitted by the Mark II plant TS (120°F) with the reactor at operating pressure. The LaSalle units also have the same TS limit for suppression pool that would require the reactor to be depressurized. Therefore, the staff agrees with the licensee that the SRV air clearing load resulting from the proposed increase of normal operational pool temperature from 100°F to 105°F will be bounded by the limiting SRV air clearing load for the LaSalle units.

(b) SRV Steam Condensation Loads

The licensee referred to GE Topical Report NEDO-30832, "Elimination of Limit on BWR Suppression Pool Temperature for SRV Discharge with Quenchers" submitted to the NRC by the BWR Owner's Group in March 1985. This report had concluded that the local pool temperature limits for the suppression pool to ensure steam condensation under stable conditions during SRV steam discharge into the pool specified in NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments" dated November 1981," could be eliminated for BWRs that utilize T or X-quencher devices. GE concluded the above, based on their findings (tabulated in the NEDO-30832 report) that the SRV condensation loads with the above quencher devices were low in comparison with other loads (e.g., SRV air clearing loads) considered in containment structural evaluations. The staff has not yet completed its evaluation of the above report. Therefore, for this Safety Evaluation, the staff has used the criteria for local pool temperature limit during SRV steam discharge into the pool that is identified in NUREG-0783

to assess whether the peak local pool temperature resulting from the proposed initial pool temperature of 105°F is acceptable. LaSalle T-quenchers have a submergence of 21.5 feet of water corresponding to a local pressure of 24 psia. The saturation temperature at 24 psia is 237.8°F. Thus, the 20°F subcooling limit identified in NUREG-0783 corresponds to a suppression pool local temperature limit of 217.8°F. The current peak local pool temperature for LaSalle is 201.3°F. The staff has performed the calculations to determine the impact of increased initial suppression pool temperature on the peak calculated local pool temperature in accordance with NUREG-0783 (Appendix A). The results indicate that a 10°F increase in the initial pool temperature will result in a increase in peak pool temperature of 5°F.

Based on the above considerations, the staff has determined that the proposed increase of operational pool temperature by 5°F will not result in a peak pool local temperature higher than the estimated allowable limit of 217.8°F. Therefore, the staff concludes that there is reasonable assurance that the proposed normal operational pool temperature limit of 105°F will not compromise the ability of the suppression pool to condense steam under stable conditions during SRV discharge of steam into the pool and, therefore, meets the intent and purpose of NUREG-0783. In this context, the staff notes that the proposed TS changes will not alter the existing requirements for (1) pool cooling whenever the pool temperature exceeds 105°F, (2) scramming the reactor whenever the pool temperature exceeds 110°F, and (3) depressurizing the reactor whenever the pool temperature exceeds 120°F.

On the basis of its review of this matter, the staff concludes that the above change to the TS for LaSalle County Station, Units 1 and 2 are acceptable.

2.3 ECCS Performance

The core cooling capability of the ECCS pumps is determined by the ability to keep the peak clad temperature of the fuel to less than 2200°F for all postulated loss of coolant accident (LOCA) events, considering an arbitrary single failure. For the LaSalle units, the most limiting LOCA event is the large recirculation line break coupled with a single failure of the LPCS diesel generator. For this event, the HPCS and two LPCI pumps are the effective ECCS pumps for core cooling.

The GE report (EAS-19-0388) presented the results of an ECCS analysis using 110°F as the initial pool temperature instead of the 100°F used in the original ECCS calculations. The results indicate that there is no significant impact on the LOCA analysis. Thus, the proposed TS change would not adversely affect ECCS performance.

On the basis of the GE information, the staff concludes that ECCS performance will remain within the limits set by 10 CFR 50, Appendix K, and thus is acceptable.

2.4. NPSH For Safety System Pumps

In accordance with Regulatory Guide 1.1, it is required that the RHR and core spray pumps have adequate net positive suction head (NPSH) without dependence on positive containment pressure during the worst case LOCA with a single failure.

The initial NPSH calculations for the LaSalle units were performed using an initial suppression pool water temperature of 100°F and assuming that all the energy in the reactor pressure vessel was absorbed by the suppression pool water following a LOCA. Using these and other assumptions, the peak suppression pool temperature was calculated to be 200°F. At that time, the NPSH margins for both the RHR pumps and the core spray pumps were determined to be adequate.

The GE report (EAS-19-0388) presents the results of a re-analysis using all of the assumptions of the initial analysis except that the initial pool temperature was assumed to be 110°F and realistic energy source terms were used. The energy input to the suppression pool was taken to be the blowdown energy from the LOCA plus decay heat calculated using the May-Witt decay heat correlation, which includes a 10% factor for conservatism. The energy input also was calculated using the 1979 ANS decay heat correlation which represents the best estimate decay heat correlation, and results in a calculated peak pool temperature of about 200°F.

Using the revised assumptions and the May-Witt decay heat correlation, GE calculated that the maximum suppression pool temperature would be approximately 212°F which would still result in adequate NPSH for the RHR pumps.

The GE report concludes that, based on the actual NPSH requirements for the core spray pumps at high water temperatures and the required mode of pump operation, the increase in initial pool temperature will still result in adequate NPSH for the core spray pumps.

Based on the GE report, and noting the conservatism built into the May-Witt correlation plus the fact that the calculation was run using 110°F rather than the proposed 100°F as the initial pool temperature, the staff concludes that the RHR and core spray pumps will have adequate NPSH.

The staff therefore concludes that the increase in suppression pool temperature requested by the licensee would not have an adverse impact upon the operation of the safety system pumps.

2.5 Emergency Operating Procedures (EOPs)

The GE report points out, correctly, that the proposed change in the suppression pool temperature limit would result in some needed changes to the EOPs. However, the staff is not now reviewing the adequacy of EOPs prior to implementation. Thus, this SER does not address changes to the EOPs. The staff expects that any change to the EOPs required as a result of this proposed change will be incorporated as a part of the ongoing EOP revision, which will be subject to later staff inspection for adequacy.

2.6 ATWS Evaluation

The TS for each of the LaSalle units now require that the reactor be scrammed by placing the mode switch in the Shutdown position whenever the suppression pool temperature exceeds 110°F. This TS requirement is not changed as a result of the requested TS amendment. Therefore, we conclude that the proposed change has no impact on the ATWS evaluation.

2.7 Summary

In summary, the staff has examined the impacts of the proposed TS changes on (1) LOCA-related containment loads, (2) safety-relief valve (SRV) operational loads, (3) ECCS performance calculations, (4) NPSH for safety system pumps, (5) Emergency Operating Procedures, and (6) ATWS evaluation, and has concluded that the proposed changes are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes to the use of the facility components located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amounts and no significant changes in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational exposure. The staff has previously determined that the amendments involve no significant hazards consideration, and there has been no public comment on such finding. 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 these amendments.

4.0 CONCLUSION

On the basis of the considerations discusses above, the staff concludes 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

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