June 3, 1999

Mr. Oliver D. Kingsley, President
Nuclear Generation Group
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M97497 AND M97498)

Dear Mr. Kingsley:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 133 to Facility Operating License No. NPF-11 and Amendment No. 118 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 December 2, 1996, as supplemented on May 27, 1999.

The amendments revise Technical Specification 3/4.4.2 to reduce the number of required Safety/Relief valves (SRVs). This change supports a modification to remove five of the currently installed SRVs due to excess capacity and to reduce the amount of valve maintenance and associated worker radiation dose. The revised TS requires that 12 of the remaining installed 13 SRVs be operable.

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
Donna Skay, Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-373, 50-374

Enclosures: 1. Amendment No. 133 to NPF-11
2. Amendment No. 118 to NPF-18
3. Safety Evaluation

cc w/encl: see next page

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*Safety evaluations incorporated with no significant changes
** See previous concurrence

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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. 133
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 December 2, 1996, as supplemented on May 27, 1999, 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) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 133, 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 license amendment is effective as of the date of its issuance and shall be implemented prior to the startup of L1C10.

FOR THE NUCLEAR REGULATORY COMMISSION

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 3, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 133

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 4-5	3/4 4-5
B 3/4 4-2	B 3/4 4-2

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of 12 of the below listed 13 reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift setting*#; all installed valves shall be closed with OPERABLE position indication.

a. 2 safety/relief valves @1205 psig ±3%
b. 3 safety/relief valves @1195 psig ±3%
c. 2 safety/relief valves @1185 psig ±3%
d. 4 safety/relief valves @1175 psig ±3%
e. 2 safety/relief valves @1150 psig ±3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more of the above required safety/relief valve stem position indicators inoperable, restore the inoperable stem position indicators to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1 The safety/relief valve stem position indicators of each safety/relief valve shall be demonstrated OPERABLE by performance of a:

a. CHANNEL CHECK at least once per 31 days, and a
b. CHANNEL CALIBRATION at least once per 18 months.**

4.4.2.2 The low-low set function shall be demonstrated not to interfere with the OPERABILITY of the safety/relief valves or the ADS by performance of a CHANNEL CALIBRATION at least once per 18 months.

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. Following testing, lift settings shall be within ±1%.
#Up to two inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until the next refueling outage.
**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

BASES

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. Analysis has shown that with the safety function of one of the thirteen safety/relief valves inoperable the reactor pressure is limited to within ASME III allowable values for the worst case upset transient. Therefore, operation with any 12 SRV's capable of opening is allowable, although all installed SRV's must be closed and have position indication to ensure that integrity of the primary coolant boundary is known to exist at all times.

Demonstration of the safety/relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the higher limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so high concentrations of chlorides are not considered harmful during these periods.

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. 118
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 December 2, 1996, as supplemented on May 27, 1999, 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) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No.118 , 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 license amendment is effective as of the date of its issuance and shall be implemented prior to the startup of L2C9.

FOR THE NUCLEAR REGULATORY COMMISSION

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 3, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 118

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 4-6	3/4 4-6
B 3/4 4-1a	3/4 4-1a

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of 12 of the below listed 13 reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift setting*#; all installed valves shall be closed with OPERABLE position indication.

a. 2 safety/relief valves @1205 psig ±3%
b. 3 safety/relief valves @1195 psig ±3%
c. 2 safety/relief valves @1185 psig ±3%
d. 4 safety/relief valves @1175 psig ±3%
e. 2 safety/relief valves @1150 psig ±3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more of the above required safety/relief valve stem position indicators inoperable, restore the inoperable stem position indicators to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1 The safety/relief valve stem position indicators of each safety/relief valve shall be demonstrated OPERABLE by performance of a:

a. CHANNEL CHECK at least once per 31 days, and a
b. CHANNEL CALIBRATION at least once per 18 months.**

4.4.2.2 The low low set function shall be demonstrated not to interfere with the OPERABILITY of the safety/relief valves or the ADS by performance of a CHANNEL CALIBRATION at least once per 18 months.

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. Following testing, lift settings shall be within ±1%.
#Up to two inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until the next refueling outage.
**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. Analysis has shown that with the safety function of one of the thirteen safety/relief valves inoperable the reactor pressure is limited to within ASME III allowable values for the worst case upset transient. Therefore, operation with any 12 SRV's capable of opening is allowable, although all installed SRV's must be closed and have position indication to ensure that integrity of the primary coolant boundary is known to exist at all times.

Demonstration of the safety/relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.133 TO FACILITY OPERATING LICENSE NO. NPF-11

AND AMENDMENT NO. 118 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 December 2, 1996, as supplemented on May 27, 1999, Commonwealth Edison Company (ComEd, the licensee) submitted proposed changes to Technical Specification (TS) Section 3/4.4.2 for LaSalle County Station, Units 1 and 2. The proposed changes will permit permanent removal of five main steam safety/relief valves (SRVs). The current plant configuration includes 18 SRVs, of which seven are also used for the automatic depressurization system (ADS).

In support of the removals, the licensee submitted an overpressurization analysis, a review of the loss-of-coolant accident (LOCA) containment response, an evaluation of high pressure systems performance, an analysis of an anticipated transient without scram (ATWS), an evaluation of SRV reliability, and an analysis of the revised piping loads. The May 27, 1999, submittal provided additional information that did not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

The safety objective of the SRVs is to prevent overpressurization of the nuclear system. This protects the nuclear system process barrier from failure which could result in the uncontrolled release of fission products. The pressure relief system at LaSalle includes eighteen SRVs arranged in five setpoint groupings: the first group of SRVs (4) set at 1205 psig, the second group of SRVs (4) set at 1195 psig, the third group of SRVs (4) set at 1185 psig, the fourth group of SRVs (4) set at 1175 psig and the fifth group of SRVs (2) set at 1150 psig. Existing TS provide a ±3% as-found tolerance and ±1% as-left setpoint tolerance.

The licensee has proposed to remove five SRVs to take advantage of the excess steam relief capacity at LaSalle. The removal of the SRVs will also: (1) reduce worker radiation exposure associated with valve maintenance; (2) reduce the likelihood of inadvertent valve opening or leakage; and (3) reduce maintenance costs. The SRVs which are used for the ADS and Low Set Logic are not affected. The two SRVs with the lowest setpoints are also not affected. Two valves from the first group, one valve from the second group and two valves from the third group are to be removed.

The licensee's submittal also included an evaluation of increased SRV setpoint tolerance from +1 percent/-3 percent to ±3 percent. The staff had previously approved the SRV setpoint tolerance increase in Amendment No. 89 (April 25, 1995) for LaSalle, Unit 2, and Amendment No. 108 (January 3, 1996) for Unit 1. Because the SRV setpoint tolerance increase has been approved, this safety evaluation will address only the removal of the SRVs.

At present, LaSalle, Unit 1, is in cycle 8 and the core is loaded with General Electric (GE) fuel only. The Unit 2 core, and future Unit 1 cores, will contain fuel supplied by Siemens Power Corporation (SPC). The analyses performed in support of SRV removal assume that the cores contain GE fuel. However, these serve only as sensitivity studies. The plant specific analysis required for each cycle's core operating limit report (COLR) will include overpressurization analysis, transient analysis and LOCA analysis which will consider the different fuel type and the reduced steam relief valve capacity.

3.0 EVALUATION

3.1. Transient Analysis/Reload Methodology

The licensee has evaluated the impact of the SRV removals on the minimum critical power ratio (MCPR) and concluded that there is no adverse safety impact due the proposed change. The cycle-specific operating limit MCPR will be based on reload transient analyses that explicitly account for SRV removal for the licensed power to flow map. Results of these analyses will be documented in the cycle-specific reload licensing reports through the normal reload process. This is acceptable to the staff.

3.2. Analysis of the Design Basis Overpressurization Event

The licensee reevaluated the limiting design basis pressurization transient using the reduced steam relief capacity limit to confirm that the vessel pressure does not exceed the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) upset limit. The ASME Code, Section III, permits pressure transients up to 10 percent over design pressure (110 percent x 1250 psig = 1375 psig). The limiting pressurization transient analyzed is a Main Steam Isolation Valve (MSIV) closure event occurring at the end of full power life without credit for a reactor trip on MSIV position sensing. The licensee analyzed the MSIV closure event (taking credit for only 10 SRVs out of the remaining 13 SRVs) using the staff approved model ODYN with the 3 percent tolerance and calculated the maximum vessel pressure to be 1341 psig. This is within the 1375 psig ASME limit, and is acceptable to the staff.

3.3. Reevaluation of the Performance of High Pressure Systems

The licensee also evaluated the performance of high pressure systems (pump capacity, discharge pressure, etc.) considering 13 SRVs and the 3 percent tolerance limit. LaSalle, Units 1 and 2, have three systems which are required to inject to the vessel at high pressure conditions: High Pressure Core Spray (HPCS), Reactor Core Isolation Cooling (RCIC) and Standby Liquid Control system (SLCS).

In order for HPCS and RCIC to meet their design requirements, the current setpoint for the lowest setpoint group of SRVs must be maintained. Because none of the lowest setpoint SRVs (group 5) are being eliminated, the maximum injection pressure for these systems is unchanged and there is no affect on HPCS and RCIC operation. The removal of SRVs will also not impact the performance of the SLCS since the maximum system pressure is based on the upper analytical pressure for the highest valve group. The valve setpoints are not changed for any group, hence there is no impact on SLCS system operation.

3.4 Evaluation of Motor-Operated Valves and Piping

In the Safety Evaluations related to Amendment Nos. 89 and 108 for LaSalle, Units 2 and 1, respectively, the staff evaluated the effects of increased differential pressure on the performance of motor-operated valves (MOVs) due to removing up to eight SRVs from service, and found it to be acceptable. The removal of eight SRVs bounds the effects of the proposed removal of five SRVs regarding the effects of differential pressures; therefore, the staff finds that the previous conclusions regarding MOV performance remain valid.

In its December 2, 1996, submittal, the licensee provided the results of its analysis of the effects of removing the proposed five specific SRVs on the main steam and SRV discharge piping. The licensee stated that, for each removed SRV, a blind flange will be installed at the SRV main steamline header connection and that the discharge piping will be partially cut back and resupported. A blind flange will also be installed at each abandoned tail pipe at the removed SRV end. The licensee stated that the modified main steamline headers will continue to meet the same classification criteria as those applied previous to the modification and are adequate to withstand the revised design basis analysis loadings. The licensee evaluated the four main steamlines and the remaining functional SRV discharge lines, supports, and supporting structures and determined the applicable loads to be acceptable in accordance with the Updated Final Safety Analysis Report (UFSAR) criteria. As a result of the piping stress analysis, the licensee determined that additional inservice inspection of some main steam system piping welds is required. These inspections will be added to the plant inservice inspection program, which is acceptable to the staff. The licensee also evaluated the SRV discharge lines to be abandoned and the associated supports and supporting structures and determined them to be acceptable for the applicable loads in accordance with the UFSAR criteria. However, the licensee determined that for the cantilevered pipe ends (i.e., the ends at the removed SRVs) of the abandoned discharge lines, the resulting deflections for the faulted conditions would need to be assessed to determine if such deflections can be accommodated without impacting any components in their vicinity. The licensee stated that, if necessary, either new supports will be added or the cantilever length will be reduced to lessen the deflections to acceptable limits. In discussions with the staff, the licensee stated it would perform this assessment and make any necessary modifications prior to plant startup following SRV removal. The staff finds that the licensee's evaluation of the main steam and SRV discharge piping meets the plant licensing basis criteria and is acceptable. The licensee further stated that the abandoned discharge piping may be eventually removed during some future outage. This is also acceptable to the staff.

3.5. Containment Response/Hydrodynamic Loads

3.5.1 Containment Response

The limiting LOCA for the peak containment temperature response for LaSalle is the design basis large-break LOCA double-ended guillotine break of the steamline. The limiting LOCA for the peak containment pressure and peak suppression pool temperature is the recirculation line break. The vessel depressurizes without any SRV actuation for both of these events. The reduction of six SRVs (five SRVs removed and one SRV assumed out-of-service) will have no impact on the peak containment pressure and temperature calculations for large-break LOCAs.

Small steamline breaks can result in high drywell temperature conditions for long periods of time as the vessel remains at high pressure for a longer time. A study performed by the licensee, with the increase in the SRV safety mode pressure setpoint and the reduction in the number of SRVs, showed an increase in the peak vessel transient pressure response and resulted in a slight increase in the initial mass and energy release to the drywell during the first actuation. However, the total mass and the energy available for release are constant and not affected by changes to the SRVs. The change to the integrated mass and energy release to the drywell up to and past the time of the peak drywell temperature and pressure is negligible. The reduction of six SRVs will have a negligible impact on the peak containment pressure and temperature calculations for small steamline breaks.

Local suppression pool temperatures were evaluated by the licensee using a worst-case scenario of six adjacent SRVs not functioning. The actual SRVs identified for removal will not result in this worst-case scenario. A slight increase in local temperatures would occur early in the event when the valves first blow down while the suppression pool temperature is colder. This condition would not continue because the LaSalle valves actuate on low-low set and only one SRV would continue to actuate. The valves proposed for elimination do not include the current low-low set valves. The reduction of six SRVs will have a negligible impact on the local suppression pool temperatures.

3.5.2 Hydrodynamic Loads

LOCA hydrodynamic loads, such as pool swell, condensation oscillation and chugging, are dependent on the containment pressure and temperature response during the LOCA. Since these are not affected by the removal of the SRVs, these LOCA hydrodynamic loads are also unaffected.

The potential impact on the asymmetric hydrodynamic load distribution in the suppression pool for a worst case condition where six adjacent SRVs are removed was assessed by the licensee. The actual SRVs identified for removal will not result in this worst-case scenario. The design basis asymmetric loads evaluated for LaSalle include a conservative case for the discharge of three adjacent SRVs. The only way the reduction of six SRVs could result in an asymmetric load which would exceed this design load would be to postulate that the remaining 12 valves discharge to form 12 bubbles simultaneously and in phase. The staff agrees with the licensee that this is not considered to be a credible event. Therefore, the current asymmetric hydrodynamic loading case for three adjacent SRVs discharging into the suppression pool remains valid.

The determination of the submerged pool boundary and submerged structure loads are based on the SRV opening pressure and are not affected by the proposed removal of the five SRVs.

3.5.3 Conclusion

The staff has reviewed the licensee's evaluation of the containment response based on a reduction of six SRVs (five removed and one SRV assumed out-of-service). The peak calculated containment temperature and pressure are not affected. The effect on the local suppression pool temperature is negligible. Therefore, the proposal to remove five SRVs is acceptable.

The staff has reviewed the licensee's evaluation of the SRV dynamic loads based on a reduction of six SRVs. LOCA hydrodynamic loads for pool swell, condensation oscillation and chugging are dependent on the current asymmetric hydrodynamic loading case for three adjacent SRVs discharging into the suppression pool remain valid. The submerged pool boundary and submerged structure loads are based on the SRV opening pressure and are not affected. Therefore, the proposal to remove five SRVs is acceptable.

3.6. Impact on Emergency Core Cooling System Response

The licensee reviewed the LOCA analysis for LaSalle, Units 1 and 2, to determine the effect of removing five SRVs on Emergency Core Cooling System (ECCS) performance. The licensee concluded that the limiting break LOCA, which is the double-ended recirculation suction line break, is not affected because the SRVs do not actuate in this scenario. For other breaks, the ADS is assumed to operate along with other ECCS functions to mitigate the consequences of the accident. Because none of the seven ADS valves are being removed by this amendment, there is no impact on ECCS performance due to SRV removal. The licensee determined that the acceptance criteria given in 10 CFR 50.46 "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors" are still satisfied for all break sizes and locations. The staff has determined that the removal of the five SRVs specified in the licensee's submittal will not impact ECCS performance in the event of a LOCA.

3.7. Anticipated Transient Without Scram (ATWS)

The licensee reevaluated the Main Steam Isolation Valve closure under anticipated transient without scram (ATWS) conditions to support the proposed SRV removals. The licensee performed the analysis using the ODYN code which was approved by the staff as discussed in a safety evaluation dated November 17, 1998. The analysis assumed credit for only 12 of the proposed remaining 13 SRVs. The results show that the vessel pressure reaches a maximum of 1378 psig which is within the ASME vessel overpressure criterion of 1500 psig for ATWS events. The long-term effect on suppression pool temperature due to the removal of five SRVs is negligible because there is little change in the total energy discharged to the pool. This is acceptable.

GE performed a review of the impact on the ATWS analysis with respect to alternate characteristic of the Siemens fuel compared with the GE fuel. GE concluded that the changes in core characteristic are small and do not have a significant impact on the ATWS analysis results. The staff finds this acceptable.

3.8. Emergency Operating Procedures (EOPs)

The EOPs instruct operators to open SRVs based on alphabetical order. The licensee has evaluated the SRV removals to ensure that no gross asymmetries will occur using this process and finds its continued applicability acceptable. This is acceptable.

3.9 Staff Audit Calculations

The staff performed analyses to assess the impact of removing 5 of the available 18 SRVs at LaSalle, Units 1 and 2. The analyses were based upon end of Cycle 7 (EOC7) conditions provided by the licensee. A TRAC_ BF1/MOD1 model was developed using cross sections for GE9B fuel developed by the staff. The cross sections were developed using the EOC7 burnup and axial power distribution information from the LaSalle, Unit 2, process computer.

The staff ran two MSIV closure ATWS cases; one with the full complement of SRVs available and one with 5 SRVs disabled. The MSIV closure ATWS was chosen because it leads to the largest vessel pressure increase and it was felt that any effects of the reduction in SRVs would manifest themselves in the MSIV closure ATWS results. The results are summarized below.

Results of MSIV Closure ATWS Cases

Case	Energy	Vessel Bottom Head Press
MSIV ATWS w/ 18 valves	57202.39 MJ	1272.9 psi
MSIV ATWS w/ 13 valves	57463.38 MJ	1287.8 psi

The above results confirm that the effect of the reduction in the number of valves is minimal and the calculated vessel bottom head pressure is less than 1500 psig, the acceptance criterion for ATWS.

3.10 Technical Specification Changes

Technical Specification Limiting Condition for Operation (LCO) 3.4.2 lists the SRVs and their required lift settings. The licensee proposes to reduce the number of SRVs in groups a, b and c from 4 in each group to 2, 3 and 2, respectively. This change reflects the five removed SRVs and is acceptable.

Technical Specification Bases Section 3/4 4.2 discusses the function of the SRVs. The licensee proposes to change the word "eighteen" to "thirteen" and replace the statement that operation with any 17 SRVs capable of opening is allowable to 12 SRVs. These changes accurately reflect the proposed configuration of 13 SRVs instead of 18 and, hence, are acceptable.

3.11 Conclusion

The proposed amendment will allow the licensee to remove a maximum of five SRVs permanently. In support of the deletions, the licensee has submitted an analysis of the limiting pressurization transient, an analysis of anticipated operational transients for impact on MCPR, a review of the LOCA analysis, a review of the ATWS event, containment response, effect on high pressure safety systems, and an evaluation of the revised piping loads. In addition, the staff performed an independent analysis of the MSIV ATWS event that confirmed that vessel pressures will not exceed allowables. The licensee will continue to ensure that acceptance criteria for the limiting pressurization transient, abnormal operational transients, and design-basis accidents will be observed. Therefore, the proposed changes are acceptable.

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. 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 (62 FR 4343). 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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