April 25, 1995

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

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M91926)

Dear Mr. Farrar:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 89 to Facility Operating License No. NPF-18 for the LaSalle County Station, Unit 2. The amendment is in response to your application dated March 31, 1995.

The amendment revises the safety/relief valve (SRV) safety function lift setting allowable tolerance band from -3/+1% to ±3% and includes a requirement for the lift settings to be within ±1% of the technical specification limit following testing.

This amendment was processed in accordance with the exigent provisions of 10 CFR 50.91(a)(6) which include a shortened public comment period of 15 days following publication of the notice of consideration of issuance in the Federal Register.

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

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

Docket No. 50-374

Enclosures: 1. Amendment No. 89 to NPF-18
2. Safety Evaluation

cc w/encls: see next page

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S.E.

D. L. Farrar
Commonwealth Edison Company

LaSalle County Station
Unit Nos. 1 and 2

cc:

Phillip P. Steptoe, Esquire
Sidley and Austin
One First National Plaza
Chicago, Illinois 60603

Assistant Attorney General
100 West Randolph Street
Suite 12
Chicago, Illinois 60601

U.S. Nuclear Regulatory Commission
Resident Inspectors Office LaSalle Station
2605 N. 21st Road
Marseilles, Illinois 61341-9756

Chairman
LaSalle County Board of Supervisors
LaSalle County Courthouse
Ottawa, Illinois 61350

Attorney General
500 South Second Street
Springfield, Illinois 62701

Chairman
Illinois Commerce Commission
Leland Building
527 East Capitol Avenue
Springfield, Illinois 62706

Illinois Department of Nuclear Safety
Office of Nuclear Facility Safety
1035 Outer Park Drive
Springfield, Illinois 62704

Regional Administrator
U.S. NRC, Region III
801 Warrenville Road
Lisle, Illinois 60532-4351

LaSalle Station Manager
LaSalle County Station
Rural Route 1
P.O. Box 220
Marseilles, Illinois 61341

Robert Cushing
Chief, Public Utilities Division
Illinois Attorney General's Office
100 West Randolph Street
Chicago, Illinois 60601

Michael I. Miller, Esquire
Sidley and Austin
One First National Plaza
Chicago, Illinois 60690

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. 89
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 March 31, 1995, 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. 89 , 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 and shall be implemented prior to the restart of Unit 2 from its sixth refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

William D. Reckley

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

Attachment:
Changes to the Technical
Specifications 89

Date of Issuance: April 25, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 89

FACILITY OPERATING LICENSE NO. NPF-18

DOCKET NO. 50-374

Replace the following page of the Appendix "A" Technical Specifications with the enclosed page. The revised page is identified by an amendment number and contains a vertical line indicating the area of change.

REMOVE	INSERT
3/4 4-6	3/4 4-6

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of 17 of the below listed 18 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. 4 safety/relief valves @1205 psig ±3%
b. 4 safety/relief valves @1195 psig ±3%
c. 4 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 safety/relief valves stuck open, provided that suppression pool average water temperature is less than 110°F, close the stuck open relief valve(s); if unable to close the open valve(s) within 2 minutes or if suppression pool average water temperature is 110°F or greater, place the reactor mode switch in the Shutdown position.

c. 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 89 TO FACILITY OPERATING LICENSE NO. NPF-18

COMMONWEALTH EDISON COMPANY

LASALLE COUNTY STATION, UNIT 2

DOCKET NO. 50-374

1.0 INTRODUCTION

By letter dated March 31, 1995, Commonwealth Edison Company (ComEd, the licensee) requested an amendment to Facility Operating License No. NPF-18 for LaSalle County Station, Unit 2. The proposed amendment revises technical specification (TS) 3.4.2 related to the required lift settings for the safety/relief valves (SRV). The licensee proposes to change the SRV safety function lift setting allowable tolerance band from -3/+1% to ±3% and includes a requirement for the lift settings to be within ±1% of the TS limit following testing.

2.0 BACKGROUND

The existing TS 3/4.4.2, Safety/Relief Valves, for Unit 2 requires 17 of the 18 SRVs to be operable. Required lift settings and allowable tolerance bands of -3% to +1% of the settings are provided in the limiting condition for operation. Testing requirements for the SRVs are addressed by TS 4.0.5 which requires inservice testing of pumps and valves in accordance with Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code).

Nuclear power plant licensees have experienced difficulty in meeting the typical 1% setpoint tolerance for SRVs. The Boiling Water Reactor Owners' Group (BWROG) submitted the General Electric Company (GE) topical report NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," to provide justification for the relaxation of SRV TS lift setting tolerance bands which were more restrictive than ±3%. On March 8, 1993, the NRC staff issued a Safety Evaluation (SE) for the GE topical report. The staff's evaluation determined that it was acceptable for licensees to submit TS amendment requests to revise lift setting tolerances to ±3% provided that the setpoints for those SRVs tested were restored to ±1% prior to plant startup. The SE instructed licensees implementing the TS modifications to provide the following plant specific analyses:

1. Transient analysis, using NRC approved methods, of all abnormal operational occurrences (AOOs) as described in NEDC-31753P utilizing a ±3% setpoint tolerance for the safety mode of SRVs.

2. Analysis of the design basis overpressurization event using the 3% tolerance limit for the SRV setpoint to confirm that the vessel pressure does not exceed ASME pressure vessel code upset limits.

3. Plant specific analyses described in Items 1 and 2 should assure that the number of SRVs included in the analyses correspond to the number of valves required to be operable in the TS.

4. Re-evaluation of the performance of high pressure systems (pump capacity, discharge pressure, etc.), motor-operated valves, and vessel instrumentation and associated piping considering the 3% tolerance limit.

5. Evaluation of the ±3% tolerance on any plant specific alternate operating modes (e.g., increased core flow, extended operating domain, etc.).

6. Evaluation of the effect of the 3% tolerance limit on the containment response during loss-of-coolant accidents (LOCAs) and the hydrodynamic loads on the SRV discharge lines and containment.

Following the issuance of the staff's SE, several BWR licensees have submitted and implemented the SRV tolerance band changes.

3.0 EXIGENT CIRCUMSTANCES

On March 18, 1995, with LaSalle, Unit 2, in a shutdown condition for its sixth refueling outage, the licensee learned that two of the six SRVs tested had lift settings that were not within the current tolerance band allowed by the TSs. This resulted in three additional SRVs being tested. On March 25, 1995, the results of those three tests showed that two additional SRVs lifted at pressures slightly outside the existing tolerance band. The remaining nine SRVs are required to be tested based on the current TSs. However, as part of a longer range plan to reduce the number of SRVs and increase the allowable lift setting tolerances, the licensee had performed much of the analyses required to justify the proposed amendment request. As a result of the above, on March 27, 1995, the licensee decided to expedite the SRV lift setting TS change for LaSalle, Unit 2, because testing would involve a significant financial cost, the collection of approximately 11 person-rem of radiation exposure by plant workers, and a delay in the restart of Unit 2. The history of the safety relief valve testing at LaSalle is such that the licensee did not anticipate the immediate need for an increased tolerance band. The licensee completed the review and submitted the request on March 31, 1995.

To avoid the radiation exposures and restart delays associated with testing the remaining nine SRVs prior to the scheduled restart (as of the date of the licensee's request), the proposed amendment needed to be issued before April 22, 1995, and, therefore, the request would not afford the normal 30-day comment period. Section 50.91(a)(6) of Title 10 of the Code of Federal Regulations specifies that the Commission may, where exigent circumstances exist, allow less than 30 days for public comment. The staff finds that the licensee used its best efforts to make a timely application following the discovery of the problem and that exigent circumstances do exist and were not the result of any intentional delay on the part of the licensee.

4.0 EVALUATION

In accordance with the staff's SE related to NEDC-31753P, the licensee provided plant specific analysis related to the increase in the SRV lift setting tolerance to ±3%. The plant specific analysis submitted to justify the change in setpoint tolerances was performed with the expectation to also revise the number of required operable SRVs. However, this submittal deals only with the revision of setting tolerances for LaSalle, Unit 2. A future licensee submittal will address proposed changes to the actual number of SRVs. The assumed reduced number of SRVs is conservative with respect to the actual plant configuration and the supporting analyses for the setting tolerance changes. A reduced number of SRVs is bounding for reactor coolant system overpressure protection and the performance of individual valves and associated discharge piping.

The licensee evaluated the potential effect of increased SRV lift pressures on fuel performance limits derived from AOOs and design basis events. The increased setpoint tolerance was found to have no impact on the minimum critical power ratio (MCPR) or the LOCA analysis. During the limiting reload licensing events for LaSalle, the MCPR occurs before the actuation of the lowest SRV setpoint. For the LOCA analysis, the automatic depressurization system (ADS) function of the SRVs is assumed to operate, but this function is not affected by the increased safety function lift setting tolerance.

The licensee provided an analysis of the design basis overpressurization event, main steam isolation valve closure with reactor scram on high flux, that assumes the +3% SRV setting tolerance. Other analysis assumptions were consistent with the current licensing basis and a conservative reduced number of SRVs was included in the analysis. The analysis resulted in a peak pressure less than the ASME upset limit of 1375 psig. The anticipated transient without scram (ATWS) event was also analyzed. The analysis results demonstrated that the peak pressure was less than the ASME emergency criterion of 1500 psig.

The performance of the high pressure core spray, reactor core isolation cooling, and standby liquid control systems were evaluated. The systems' performance were found to remain capable of performing their safety functions at the increased maximum pressures associated with the +3% SRV setting tolerance. The licensee reviewed the potential impact of the increased differential pressures on valves associated with the systems and determined that the valves would perform as required.

The evaluations of the effect of the 3% tolerance limit on the containment response during LOCAs and the hydrodynamic loads on the SRV discharge lines and containment were included in the submittal. The evaluation determined that containment pressure and temperature responses to events were not impacted by the increased SRV setpoint tolerance. The increased forces on piping and structures introduced by higher flow rates associated with an increased maximum lift setting pressure was evaluated. The evaluations determined that available margins in the design calculations were sufficient to accommodate the increased loads for a +3% setpoint tolerance.

The staff concludes that a lift setting tolerance of ±3% was properly analyzed by the licensee in terms of the potential effects on plant equipment and design requirements. The conditions for plant specific analyses which were specified in the staff's SE of NEDC-31753P have been satisfied. The staff finds the proposed TS change acceptable.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The staff has evaluated the proposed amendment against the three factors as part of the determination:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The probability of an accident previously evaluated will not increase as a result of this change because the only changes are the tolerances for the SRV opening setpoints and the speed of the reactor core isolation cooling system (RCIC) turbine and pump. Changing the maximum allowable opening setpoint for the SRVs does not cause any accident previously evaluated to occur, or degrade valve or system performance in any way so as to cause an accident to occur with an increased frequency. In addition, the increased speed of the RCIC turbine and pump are within the design limits of the system. RCIC operability and failure probabilities are not impacted by this change.

The consequences of an ASME overpressurization event are not significantly increased and do not exceed the previously accepted licensing criteria for this event. General Electric has calculated the revised peak vessel pressure for LaSalle to be 1341 psig, which is well below the 1375 psig criterion of the ASME Code for upset conditions referenced in the Updated Final Safety Analysis Report (UFSAR) (Section 5.2.2, Overpressurization Protection), NUREG-0519 (Safety Evaluation Report related to the operation of LaSalle

County Station, Units 1 and 2, March 1981), and NUREG-0800 (Standard Review Plan, Section 15.2-4, Closure of Main Steam Isolation Valves (BWR)). General Electric has also performed an analysis of the limiting ATWS event, which is the main steam isolation valve (MSIV) closure event. This analysis calculated the peak vessel pressure to be 1457 psig, which is well below the 1500 psig criterion of the ASME Code for emergency conditions.

Per NUREG-0519, listed above, Section 5.4.1, and TS 4.7.3.b, the RCIC pump is required to develop flow greater than or equal to 600 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is supplied to the turbine at 1000 +20, -80 psig. Increasing the turbine and pump speed ensures these criteria will still be met and the consequences of an accident will not increase.

Therefore, there is not a significant increase in the consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The only physical changes are to increase the allowable tolerances for SRV opening setpoints and to increase the RCIC pump and turbine speeds. These changes do not result in any changed component interactions. The SRVs and RCIC will still provide the functions for which they were designed. Since all of the systems evaluated will continue to function as intended, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed change does not involve a significant reduction in the margin of safety.

While the calculated peak vessel pressures for the ASME overpressurization event and the MSIV closure ATWS event are larger than that previously calculated without the proposed setpoint tolerance increases, the new peak pressures remain far below the respective licensing acceptance limits associated with these events. These licensing acceptance limits have been previously evaluated as providing a sufficient margin of safety. For other accidents and transients, the increased setpoint tolerances have a negligible, if any, effect on the results, so the margin of safety is preserved.

Based upon the above considerations, the staff concludes that the amendment involves no significant hazards consideration.

6.0 STATE CONSULTATION

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

7.0 ENVIRONMENTAL CONSIDERATION

This amendment changes 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 changes a surveillance requirement. The NRC staff has determined that this amendment involves 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 this amendment involves no significant hazards consideration, and there has been no public comment on such finding (60 FR 17590). Accordingly, this amendment meets 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 this amendment.

8.0 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: William Reckley, NRR/DRPW

Date: April 25, 1995

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 89 TO FACILITY OPERATING LICENSE NO. NPF-18

COMMONWEALTH EDISON COMPANY

LASALLE COUNTY STATION, UNIT 2

DOCKET NO. 50-374

1.0 INTRODUCTION

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2.0 BACKGROUND

The existing TS 3/4.4.2, Safety/Relief Valves, for Unit 2 requires 17 of the 18 SRVs to be operable. Required lift settings and allowable tolerance bands of -3% to +1% of the settings are provided in the limiting condition for operation. Testing requirements for the SRVs are addressed by TS 4.0.5 which requires inservice testing of pumps and valves in accordance with Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code).

Nuclear power plant licensees have experienced difficulty in meeting the typical 1% setpoint tolerance for SRVs. The Boiling Water Reactor Owners' Group (BWROG) submitted the General Electric Company (GE) topical report NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," to provide justification for the relaxation of SRV TS lift setting tolerance bands which were more restrictive than ±3%. On March 8, 1993, the NRC staff issued a Safety Evaluation (SE) for the GE topical report. The staff's evaluation determined that it was acceptable for licensees to submit TS amendment requests to revise lift setting tolerances to ±3% provided that the setpoints for those SRVs tested were restored to ±1% prior to plant startup. The SE instructed licensees implementing the TS modifications to provide the following plant specific analyses:

1. Transient analysis, using NRC approved methods, of all abnormal operational occurrences (AOOs) as described in NEDC-31753P utilizing a ±3% setpoint tolerance for the safety mode of SRVs.

2. Analysis of the design basis overpressurization event using the 3% tolerance limit for the SRV setpoint to confirm that the vessel pressure does not exceed ASME pressure vessel code upset limits.

3. Plant specific analyses described in Items 1 and 2 should assure that the number of SRVs included in the analyses correspond to the number of valves required to be operable in the TS.

4. Re-evaluation of the performance of high pressure systems (pump capacity, discharge pressure, etc.), motor-operated valves, and vessel instrumentation and associated piping considering the 3% tolerance limit.

5. Evaluation of the ±3% tolerance on any plant specific alternate operating modes (e.g., increased core flow, extended operating domain, etc.).

6. Evaluation of the effect of the 3% tolerance limit on the containment response during loss-of-coolant accidents (LOCAs) and the hydrodynamic loads on the SRV discharge lines and containment.

Following the issuance of the staff's SE, several BWR licensees have submitted and implemented the SRV tolerance band changes.

3.0 EXIGENT CIRCUMSTANCES

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To avoid the radiation exposures and restart delays associated with testing the remaining nine SRVs prior to the scheduled restart (as of the date of the licensee's request), the proposed amendment needed to be issued before April 22, 1995, and, therefore, the request would not afford the normal 30-day comment period. Section 50.91(a)(6) of Title 10 of the Code of Federal Regulations specifies that the Commission may, where exigent circumstances exist, allow less than 30 days for public comment. The staff finds that the licensee used its best efforts to make a timely application following the discovery of the problem and that exigent circumstances do exist and were not the result of any intentional delay on the part of the licensee.

4.0 EVALUATION

In accordance with the staff's SE related to NEDC-31753P, the licensee provided plant specific analysis related to the increase in the SRV lift setting tolerance to ±3%. The plant specific analysis submitted to justify the change in setpoint tolerances was performed with the expectation to also revise the number of required operable SRVs. However, this submittal deals only with the revision of setting tolerances for LaSalle, Unit 2. A future licensee submittal will address proposed changes to the actual number of SRVs. The assumed reduced number of SRVs is conservative with respect to the actual plant configuration and the supporting analyses for the setting tolerance changes. A reduced number of SRVs is bounding for reactor coolant system overpressure protection and the performance of individual valves and associated discharge piping.

The licensee evaluated the potential effect of increased SRV lift pressures on fuel performance limits derived from AOOs and design basis events. The increased setpoint tolerance was found to have no impact on the minimum critical power ratio (MCPR) or the LOCA analysis. During the limiting reload licensing events for LaSalle, the MCPR occurs before the actuation of the lowest SRV setpoint. For the LOCA analysis, the automatic depressurization system (ADS) function of the SRVs is assumed to operate, but this function is not affected by the increased safety function lift setting tolerance.

The licensee provided an analysis of the design basis overpressurization event, main steam isolation valve closure with reactor scram on high flux, that assumes the +3% SRV setting tolerance. Other analysis assumptions were consistent with the current licensing basis and a conservative reduced number of SRVs was included in the analysis. The analysis resulted in a peak pressure less than the ASME upset limit of 1375 psig. The anticipated transient without scram (ATWS) event was also analyzed. The analysis results demonstrated that the peak pressure was less than the ASME emergency criterion of 1500 psig.

The performance of the high pressure core spray, reactor core isolation cooling, and standby liquid control systems were evaluated. The systems' performance were found to remain capable of performing their safety functions at the increased maximum pressures associated with the +3% SRV setting tolerance. The licensee reviewed the potential impact of the increased differential pressures on valves associated with the systems and determined that the valves would perform as required.

The evaluations of the effect of the 3% tolerance limit on the containment response during LOCAs and the hydrodynamic loads on the SRV discharge lines and containment were included in the submittal. The evaluation determined that containment pressure and temperature responses to events were not impacted by the increased SRV setpoint tolerance. The increased forces on piping and structures introduced by higher flow rates associated with an increased maximum lift setting pressure was evaluated. The evaluations determined that available margins in the design calculations were sufficient to accommodate the increased loads for a +3% setpoint tolerance.

The staff concludes that a lift setting tolerance of ±3% was properly analyzed by the licensee in terms of the potential effects on plant equipment and design requirements. The conditions for plant specific analyses which were specified in the staff's SE of NEDC-31753P have been satisfied. The staff finds the proposed TS change acceptable.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The staff has evaluated the proposed amendment against the three factors as part of the determination:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The probability of an accident previously evaluated will not increase as a result of this change because the only changes are the tolerances for the SRV opening setpoints and the speed of the reactor core isolation cooling system (RCIC) turbine and pump. Changing the maximum allowable opening setpoint for the SRVs does not cause any accident previously evaluated to occur, or degrade valve or system performance in any way so as to cause an accident to occur with an increased frequency. In addition, the increased speed of the RCIC turbine and pump are within the design limits of the system. RCIC operability and failure probabilities are not impacted by this change.

The consequences of an ASME overpressurization event are not significantly increased and do not exceed the previously accepted licensing criteria for this event. General Electric has calculated the revised peak vessel pressure for LaSalle to be 1341 psig, which is well below the 1375 psig criterion of the ASME Code for upset conditions referenced in the Updated Final Safety Analysis Report (UFSAR) (Section 5.2.2, Overpressurization Protection), NUREG-0519 (Safety Evaluation Report related to the operation of LaSalle

County Station, Units 1 and 2, March 1981), and NUREG-0800 (Standard Review Plan, Section 15.2-4, Closure of Main Steam Isolation Valves (BWR)). General Electric has also performed an analysis of the limiting ATWS event, which is the main steam isolation valve (MSIV) closure event. This analysis calculated the peak vessel pressure to be 1457 psig, which is well below the 1500 psig criterion of the ASME Code for emergency conditions.

Per NUREG-0519, listed above, Section 5.4.1, and TS 4.7.3.b, the RCIC pump is required to develop flow greater than or equal to 600 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is supplied to the turbine at 1000 +20, -80 psig. Increasing the turbine and pump speed ensures these criteria will still be met and the consequences of an accident will not increase.

Therefore, there is not a significant increase in the consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The only physical changes are to increase the allowable tolerances for SRV opening setpoints and to increase the RCIC pump and turbine speeds. These changes do not result in any changed component interactions. The SRVs and RCIC will still provide the functions for which they were designed. Since all of the systems evaluated will continue to function as intended, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed change does not involve a significant reduction in the margin of safety.

While the calculated peak vessel pressures for the ASME overpressurization event and the MSIV closure ATWS event are larger than that previously calculated without the proposed setpoint tolerance increases, the new peak pressures remain far below the respective licensing acceptance limits associated with these events. These licensing acceptance limits have been previously evaluated as providing a sufficient margin of safety. For other accidents and transients, the increased setpoint tolerances have a negligible, if any, effect on the results, so the margin of safety is preserved.

Based upon the above considerations, the staff concludes that the amendment involves no significant hazards consideration.

6.0 STATE CONSULTATION

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

7.0 ENVIRONMENTAL CONSIDERATION

This amendment changes 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 changes a surveillance requirement. The NRC staff has determined that this amendment involves 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 this amendment involves no significant hazards consideration, and there has been no public comment on such finding (60 FR 17590). Accordingly, this amendment meets 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 this amendment.

8.0 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: William Reckley, NRR/DRPW

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