November 8, 2000

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: LASALLE COUNTY STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS (TAC NOS. MA8403 AND MA8404)

Dear Mr. Kingsley:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 144 to Facility Operating License No. NPF-11 and Amendment No. 130 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 February 29, 2000, as supplemented by letters dated June 26 and August 18, 2000.

The amendments revise the pressure-temperature (P-T) limits for heatup, cooldown, critical operation and inservice leak and hydrostatic test limitations for the reactor pressure vessel (RPV). The amendments replace the current RPV P-T limit curves with three recalculated curves that are applicable to 32 effective full power years. The staff has approved the revised limits for an interim period not to exceed December 15, 2002.

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,

/RA/

Donna M. 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. 144 to NPF-11
2. Amendment No. 130 to NPF-18
3. Safety Evaluation

cc w/encls: See next page

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NAME	DSKAY	CMOORE	KWICHMAN*	C marco	AMENDOLA
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*input provided by memo incorporated with no significant changes

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

November 8, 2000

Mr. Oliver D. Kingsley, President
Nuclear Generation Group
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
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Donna M. Skay

Donna M. 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. 144 to NPF-11
2. Amendment No. 130 to NPF-18
3. Safety Evaluation

cc w/encls: See next page

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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. 144
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 February 29, 2000, as supplemented by letters dated June 26 and August 18, 2000, 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. 144 , 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 until December 15, 2002. This amendment shall be implemented within 30 days of issuance.

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: November 8, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 144

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. Pages indicated with an asterisk are provided for convenience.

REMOVE	INSERT
XIX	XIX
XXII	XXII
XXIII	XXIII
3/4 4-16	3/4 4-16
3/4 4-17	3/4 4-17
3/4 4-18	3/4 4-18
3/4 4-18a	3/4 4-18a
3/4 4-19	3/4 4-19
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3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 for hydrostatic or leak testing; Figure 3.4.6.1-1a for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and Figure 3.4.6.1-1b for operations with a critical core other than low power PHYSICS TESTS, with:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 100°F in any one hour period,

c A maximum temperature change of less than or equal to 20°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d The reactor vessel flange and head flange temperature greater than or equal to 72°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figures 3.4.6.1-1, 3.4.6.1-1a, and 3.4.6.1-1b, as applicable, at least once per 30 minutes.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1b within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality.

4.4.6.1.3 The reactor vessel material specimens shall be removed and examined to determine reactor pressure vessel fluence as a function of time and THERMAL POWER as required by 10 CFR Part 50, Appendix H. The results of these fluence determinations shall be used to update the curves of Figures 3.4.6.1-1, 3.4.6.1-1a, and 3.4.6.1-1b.

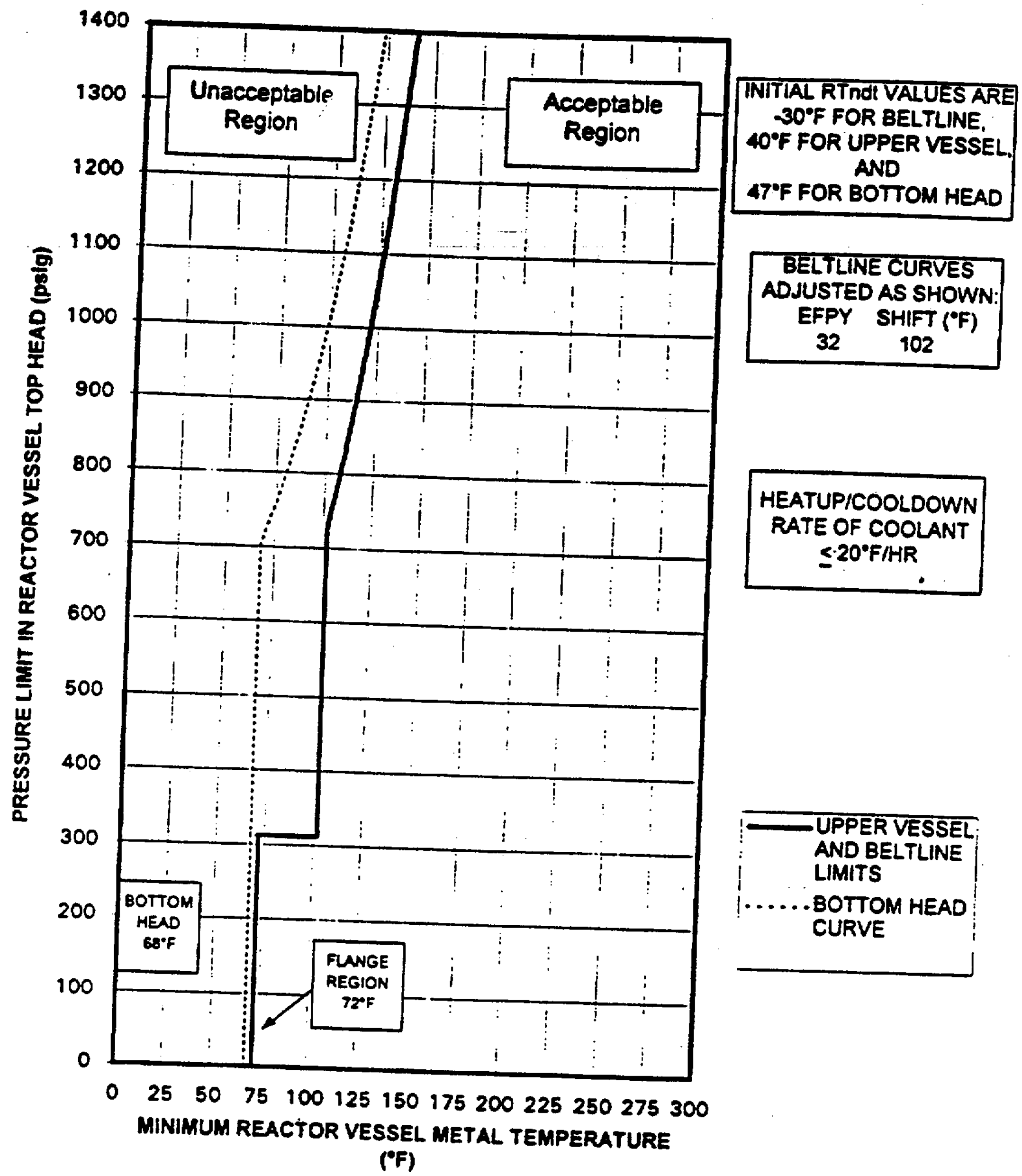
4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 72°F:

a. In OPERATIONAL CONDITION 4 when the reactor coolant temperature is:

   1. ≤92°F, at least once per 12 hours.

   2. ≤77°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

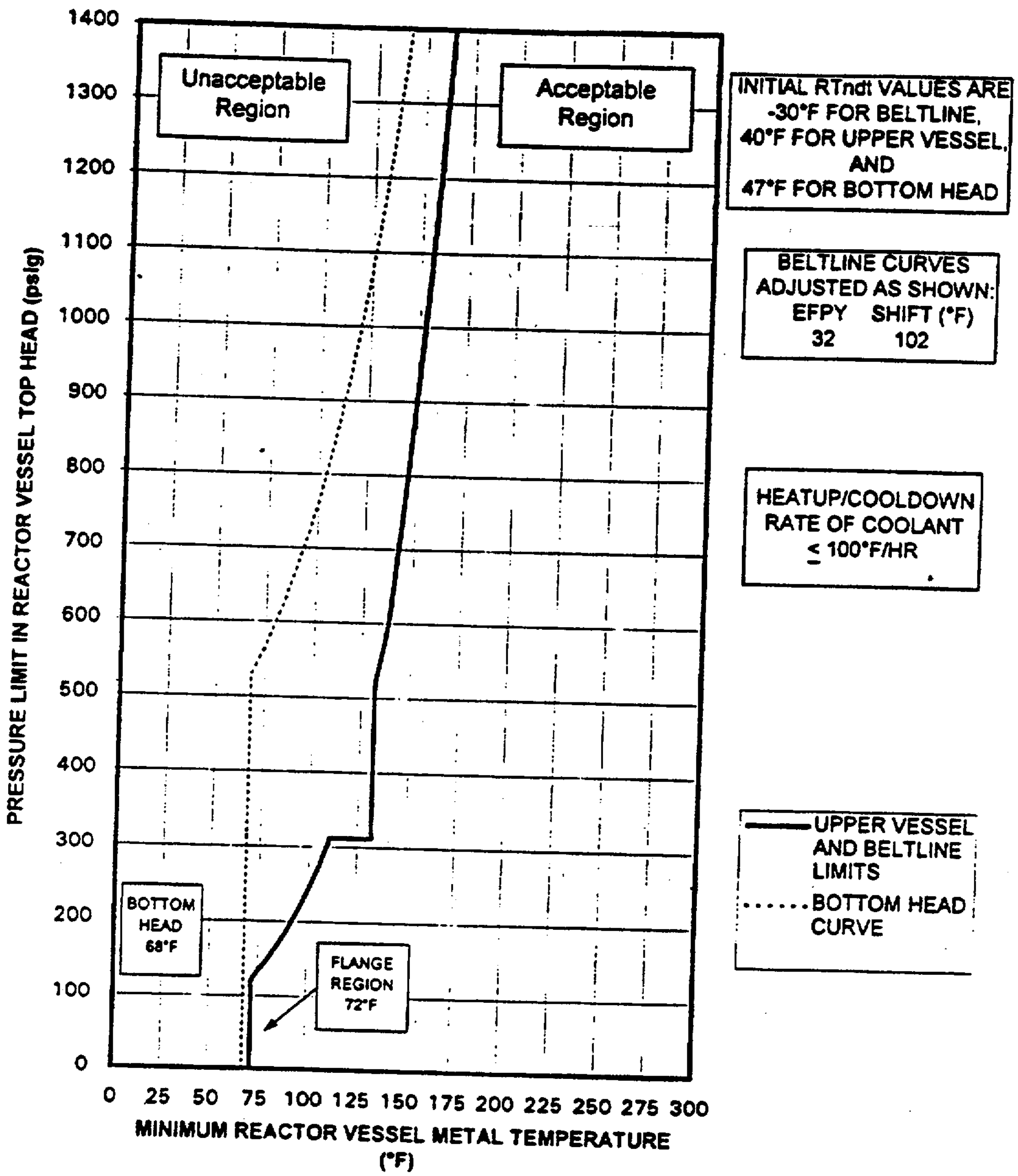


P-T Curve for Hydrostatic or Leak Testing

Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure

Valid to 32 EFPY

Figure 3.4.6.1-1

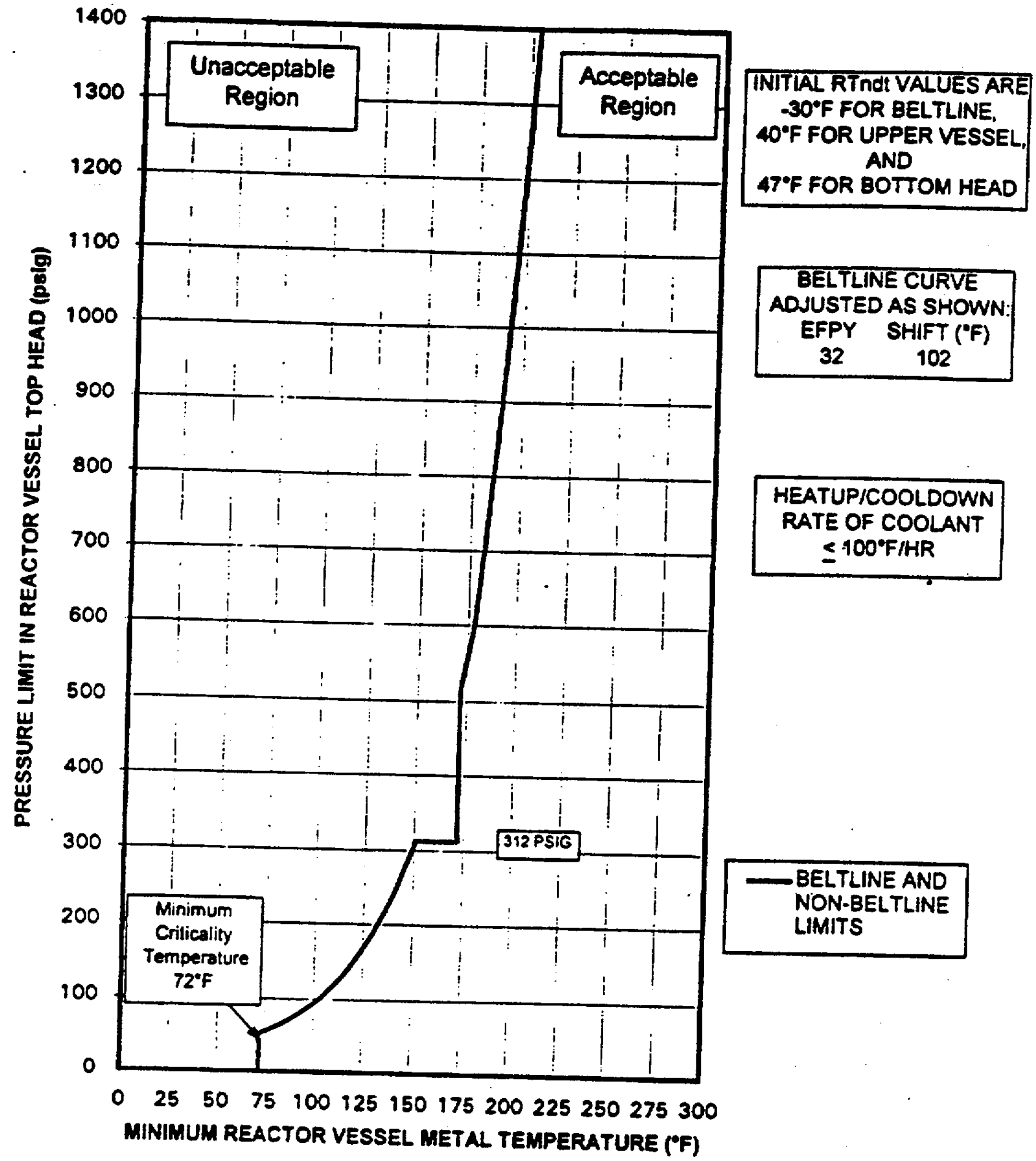


P-T Curve for Heatup by Non-Nuclear Means,
Cooldown Following a Nuclear Shutdown and Low Power PHYSICS TESTS

Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure

Valid to 32 EFPY

Figure 3.4.6.1-1a



P-T Curve for Operation with a Critical Core other than low power PHYSICS TESTS

Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure

Valid to 32 EFPY

Figure 3.4.6.1-1b

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than 1020 psig.

APPLICABILITY: OPERATIONAL CONDITION 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding 1020 psig, reduce the pressure to less than 1020 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than 1020 psig at least once per 12 hours.

*Not applicable during anticipated transients.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are provided in Section 4 of the UFSAR. Reactor operation and resultant fast neutron, E greater than 1 Mev, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, nickel content and copper content of the material in question, can be predicted using the recommendations of Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The pressure/temperature limit curve, Figures 3.4.6.1-1, 3.4.6.1-1a, and 3.4.6.1-1b, includes predicted adjustments for this shift in RT_{NDT} at the end of thirty-two effective full power years (EFPY).

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The specimen withdrawal schedule is provided in UFSAR Section 4. Since the neutron spectra at the material specimens and vessel inside radius are essentially identical, the irradiated specimens can be used with confidence in predicting

PRESSURE/TEMPERATURE LIMITS (Continued)

reactor vessel material transition temperature shift. The operating limit curves of Figures 3.4.6.1-1, 3.4.6.1-1a, and 3.4.6.1-1b, shall be adjusted, as required, on the basis of the specimen data and the recommendations of Regulatory Guide 1.99, Rev. 2.

The pressure-temperature limit lines shown in Figures 3.4.6.1-1, 3.4.6.1-1a, and 3.4.6.1-1b, for reactor criticality and for inservice leak and hydrostatic testing have been established using the requirements of Appendix G to 10 CFR Part 50.

3/4.4.7 MAIN STEAMLINE ISOLATION VALVES

Double isolation valves are provided on each of the main steamlines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda through Summer 1975.

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.4.9 RESIDUAL HEAT REMOVAL

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication; however, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.

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. 130
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 February 29, 2000, as supplemented by letters dated June 26 and August 18, 2000, 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. 130 , 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 until December 15, 2002. This amendment shall be implemented within 30 days of issuance.

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: November 8, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 130

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
XIX	XIX
XXII	XXII
XXIII	XXIII
3/4 4-17	3/4 4-17
3/4 4-18	3/4 4-18
3/4 4-19	3/4 4-19
3/4 4-19a	3/4 4-19a
3/4 4-20	3/4 4-20
B 3/4 4-4	B 3/4 4-4
B 3/4 4-5	B 3/4 4-5
B 3/4 4-6	- - -

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REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 for hydrostatic or leak testing; Figure 3.4.6.1-1a for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and Figure 3.4.6.1-1b for operations with a critical core other than low power PHYSICS TESTS, with:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 100°F in any one hour period,

c A maximum temperature change of less than or equal to 20°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d The reactor vessel flange and head flange temperature greater than or equal to 86°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figures 3.4.6.1-1, 3.4.6.1-1a, and 3.4.6.1-1b, as applicable, at least once per 30 minutes.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1b within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality.

4.4.6.1.3 The reactor vessel material specimens shall be removed and examined to determine reactor pressure vessel fluence as a function of time and THERMAL POWER as required by 10 CFR Part 50, Appendix H. The results of these fluence determinations shall be used to update the curves of Figures 3.4.6.1-1, 3.4.6.1-1a, and 3.4.6.1-1b.

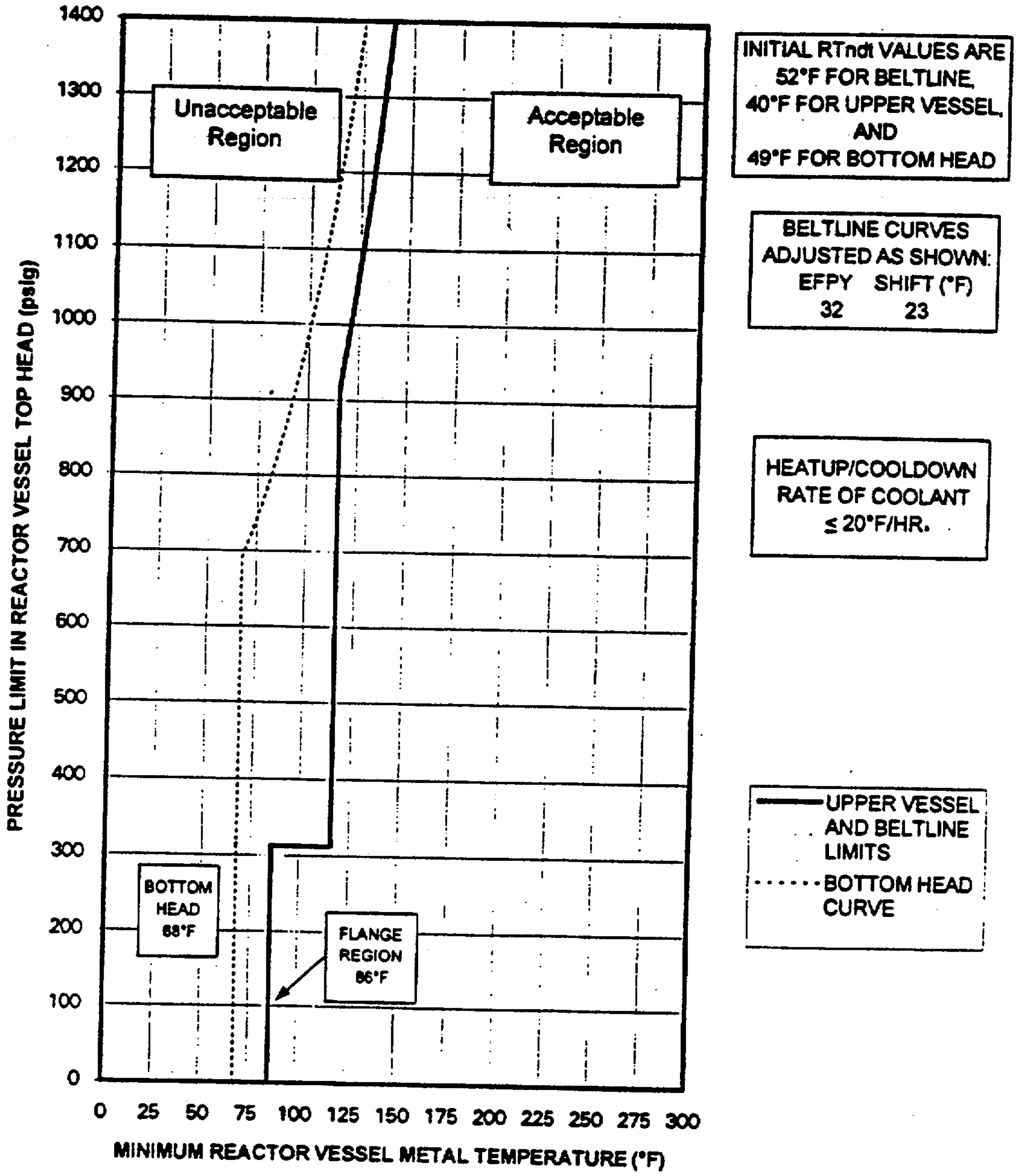
4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 86°F:

a. In OPERATIONAL CONDITION 4 when the reactor coolant temperature is:

   1. ≤106°F, at least once per 12 hours.

   2. ≤91°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

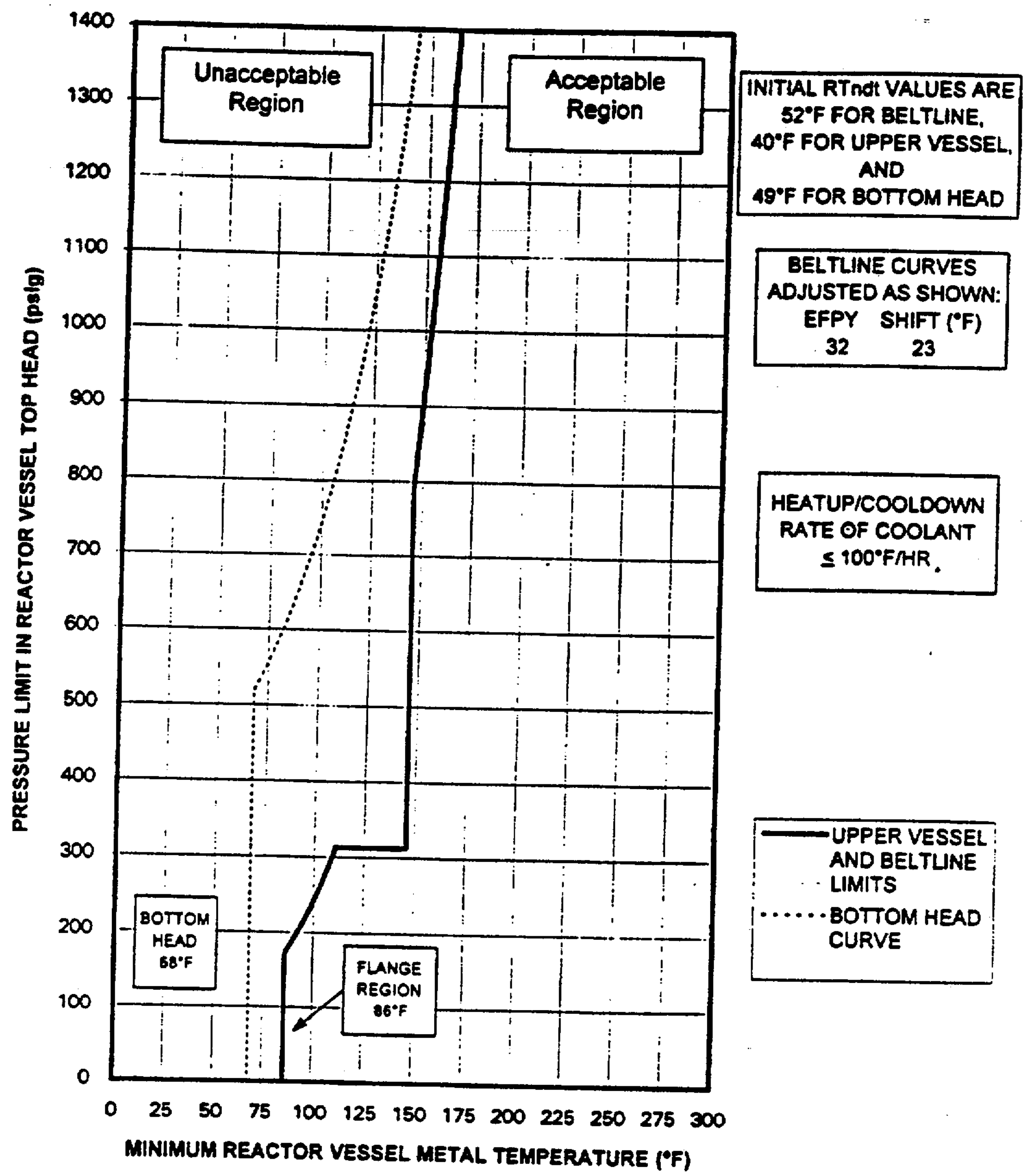


P-T Curve for Hydrostatic or Leak Testing

Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure

Valid to 32 EFPY

Figure 3.4.6.1-1

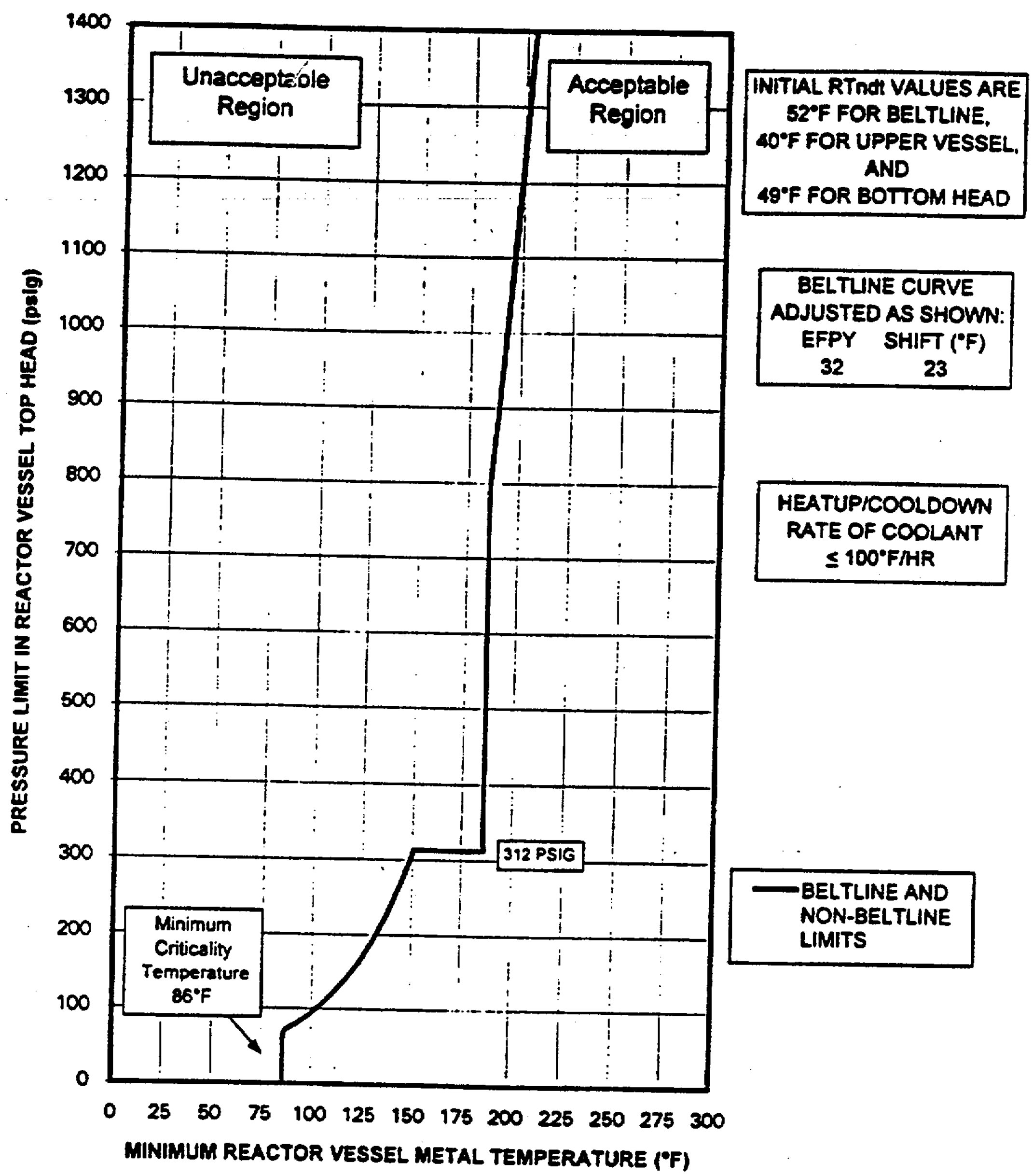


P-T Curve for Heatup by Non-Nuclear Means,
Cooldown Following a Nuclear Shutdown and Low Power PHYSICS TESTS

Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure

Valid to 32 EFPY

Figure 3.4.6.1-1a



P-T Curve for Operation with a Critical Core other than low power PHYSICS TESTS

Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure

Valid to 32 EFPY

Figure 3.4.6.1-1b

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are provided in Section 4 of the UFSAR. Reactor operation and resultant fast neutron, E greater than 1 Mev, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, nickel content and copper content of the material in question, can be predicted using the recommendations of Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The pressure/temperature limit curve, Figures 3.4.6.1-1, 3.4.6.1-1a, and 3.4.6.1-1b, includes predicted adjustments for this shift in RT_{NDT} at the end of thirty-two effective full power years (EFPY).

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR Part 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The specimen withdrawal schedule is provided in UFSAR Section 4. Since the neutron spectra at the material specimens and vessel inside radius are essentially identical, the irradiated specimens can be used with confidence in predicting

PRESSURE/TEMPERATURE LIMITS (Continued)

reactor vessel material transition temperature shift. The operating limit curves of Figures 3.4.6.1-1, 3.4.6.1-1a, and 3.4.6.1-1b, shall be adjusted, as required, on the basis of the specimen data and the recommendations of Regulatory Guide 1.99, Rev. 2.

The pressure-temperature limit lines shown in Figures 3.4.6.1-1, 3.4.6.1-1a, and 3.4.6.1-1b, for reactor criticality and for inservice leak and hydrostatic testing have been established using the requirements of Appendix G to 10 CFR Part 50.

3/4.4.7 MAIN STEAMLINE ISOLATION VALVES

Double isolation valves are provided on each of the main steamlines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda through Summer 1975.

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.4.9 RESIDUAL HEAT REMOVAL

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication; however, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 144 TO FACILITY OPERATING LICENSE NO. NPF-11

AND AMENDMENT NO. 130 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 February 29, 2000, as supplemented by letters dated June 26 and August 18, 2000, Commonwealth Edison Company (ComEd, the licensee) submitted proposed changes to the Technical Specifications (TS) for LaSalle County Station, Units 1 and 2. The proposed amendments would revise the pressure-temperature (P-T) limits for heatup, cooldown, critical operation and inservice leak and hydrostatic test limitations for the reactor pressure vessel (RPV). The amendments replace the current RPV P-T limit curves with three recalculated curves that are applicable to 32 effective full power years (EFPY). The June 26 and August 18, 2000, submittals provided additional information that did not change the scope of the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

Requirements for Generating P-T Limits for Nuclear Power Generation Facilities

The U.S. Nuclear Regulatory Commission (NRC) has established requirements in Appendix G of Part 50 to Title 10, *Code of Federal Regulations* (10 CFR Part 50, Appendix G), to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The Appendix to Part 50 requires the P-T limits for an operating plant to be at least as conservative as those that would be generated if the methods of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) (Appendix G to the Code) were applied. The methodology of Appendix G to the Code postulates the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum applied stress. For materials in the beltline and upper and lower head regions of the RPV, the maximum flaw size is postulated to have a depth that is equal to one-fourth of the thickness and a length equal to 1.5 times the thickness. For the case of evaluating RPV nozzles, the surface flaw is postulated to propagate parallel to the axis of the nozzle's corner radius. The basic parameter in Appendix G to the Code for calculating P-T limit curves is the stress intensity factor, K_I, which is a function of the stress state and flaw configuration. The methodology requires that licensees determine the reference stress intensity (K_{Ia}) factors, which vary as a function of temperature, from the reactor coolant system (RCS) operating temperatures, and from the adjusted reference temperatures (ARTs) for the limiting materials in the RPV. Thus, the critical locations in the RPV beltline and head regions are the 1/4-thickness (1/4T) and 3/4-thickness (3/4T) locations, which correspond to the points of the crack tips if the flaws are initiated and grown from the inside and outside surfaces of the vessel, respectively. Regulatory Guide (RG) 1.99, Revision 2, provides an acceptable method of calculating ARTs for ferritic RPV materials; the methods of RG 1.99, Revision 2, include methods for adjusting the ARTs of materials in the beltline region of the RPV, where the effects of neutron irradiation may induce an increased level of embrittlement in the materials.

The methodology of Appendix G of the Code requires that P-T curves must satisfy a safety factor of 2.0 on stress intensities arising from primary membrane and bending stresses during normal plant operations (including heatups, cooldowns, and transient operating conditions), and a safety factor of 1.5 on stress intensities arising from primary membrane and bending stresses when leak rate or hydrostatic pressure tests are performed on the RCS. Table 1 to 10 CFR Part 50, Appendix G, provides the staff's criteria for meeting the P-T limit requirements of Appendix G to the Code and the minimum temperature requirements of the rule for bolting up the vessel during normal and pressure testing operations.

3.0 EVALUATION

3.1 Requested Action

On February 29, 2000, as supplemented by letters dated June 26 and August 18, 2000, ComEd requested a license amendment to update the P-T limit curves for LaSalle County Station, Units 1 and 2. The extension requested corresponds to the end of the current license. The proposed fluence value for 32 EFPYs was determined by extrapolation from the value used for the current P-T curves. In response to staff concerns with the validity of the original fluence determination, the licensee proposed, in a letter dated August 18, 2000, to limit the applicability of the P-T curves to December 15, 2002. In the interim, the licensee will complete a fluence evaluation based on the recommendations of Draft Regulatory Guide (DG)-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated August 1999 (to be published by the end of 2000). By December 15, 2002, the licensee will propose revised curves for 32 EFPY.

On November 8, 2000, pursuant to 10 CFR 50.12, the NRC granted an exemption to allow ComEd to deviate from the requirements of 10 CFR Part 50, Appendix G, and to use Code Case N-640 as the bases for generating the LaSalle, Units 1 and 2, P-T limit curves effective to 22 EFPY and 32 EFPY. Approval to use Code Case N-640 allows licensees to use the lower bound static initiation fracture toughness value (K_{1c}) equation as the basis for establishing the P-T limits in lieu of using the lower bound crack arrest fracture toughness value (K_{1a}) equation, which is the method invoked by Appendix G to the Code.

3.2 Assessment of Neutron Fluence Values

The staff performed an independent review of the neutron fluence information and values submitted in ComEd's application and GE-NE-523-A166-1294 report, "LaSalle Unit 1 Reactor Pressure Vessel Surveillance Materials Testing and Analysis" dated June 1995. The staff noted from its review that the original fluence values were obtained from in-vessel surveillance capsules which were removed in the middle 1980's. The staff determined that the information submitted in the ComEd letters and report failed to establish the basis for the fluence estimate utilized by the licensee in its proposed P-T curves for LaSalle, Units 1 and 2.

Since the staff could not establish the validity of the original fluence determination for LaSalle, Units 1 and 2, ComEd proposed, by letter dated August 18, 2000, to limit the period of applicability of the LaSalle, Units 1 and 2, 32 EFPY P-T limit curves to December 15, 2002. At that time the licensee will prepare new fluence values which will comply with the requirements of the draft Regulatory Guide DG-1053. The staff reviewed whether the 32 EFPY curves are acceptable for use until December 2002.

The staff has determined that, although the basis for the estimated fluence value used by the licensee in its proposed P-T curves cannot be well established, the conservatisms incorporated in the value are sufficient to allow this value to be used until a new value can be calculated using the guidance in DG-1053. These conservatisms include: (1) the proposed fluence values have a conservatism of 61.9 percent and 63.1 percent for LaSalle, Units 1 and 2, respectively, due to the limited period of operation; (2) the surveillance capsule results were adjusted for power uprate and this added to the conservatism of the 32 EFPY estimated fluence; (3) the LaSalle units have been operating with longer fuel cycles which entail low neutron leakage loadings, thus, providing another source of conservatism; and (4) the proposed values compare favorably with the other 251-inch vessel units, thus, offer reassurance that the values are in a reasonable range.

Based on the above and considering the limited time of applicability of the LaSalle, Units 1 and 2, proposed P-T curves, the staff concludes there is reasonable assurance of safety and finds the proposed curves acceptable for use until December 15, 2002.

3.3 P-T Limit Curve Assessment

For the LaSalle, Units 1 and 2, RPVs, the licensee provided the P-T limit curves for normal operating conditions and pressure testing conditions effective to 22 EFPY and 32 EFPY. For the normal operating conditions with the core not critical and for pressure testing conditions, individual P-T curves were proposed for lower head in addition to the composite curves proposed for the beltline and nozzles regions of the RPVs. To test the validity of the licensee's proposed curves, the staff performed an independent assessment of the licensee's submittal. The staff applied the methodologies of 1995 Edition of Appendix G to the Code and 10 CFR Part 50, Appendix G, as modified by the methodologies of ASME Code Case N-640, as the bases for its independent assessment. For the evaluation of the RPV nozzles, the staff also modified the methods of Appendix G to the Code by the nozzle evaluation methods proposed in Appendix 5 of Welding Research Council Bulletin WRC-175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials" (August 1972).

The staff's assessment also included an independent calculation of the ART values for both the 1/4T and 3/4T locations of the LaSalle, Units 1 and 2, RPV beltline regions based on the neutron fluence specified in ComEd's submittal effective to 32 EFPY. For the evaluation of the limiting beltline materials, the staff confirmed that the ARTs and P-T limit curves were based on the methodology of RG 1.99, Revision 2. For the evaluation of the limiting material in the limiting nozzle and lower head evaluations, the staff applied the plant specific design basis data provided by the licensee.

The staff determined that ComEd's P-T limit methods were based on conservative assumptions that made the proposed P-T limit curves as conservative or slightly more conservative that the P-T limit curves generated by the staff. The staff also confirmed that ComEd's P-T limit curves included appropriate minimum temperature requirements that were at least as conservative as those required in Table 1 to 10 CFR Part 50, Appendix G, as exempted and modified by the Code Case methods.

4.0 CONCLUSIONS

Based on its review of ComEd's proposed P-T limit curves for LaSalle, Units 1 and 2, the staff has determined that the proposed P-T limit curves satisfy the requirements of 10 CFR 50.60(a), "Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation;" Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements;" and Appendix G to the 1995 Edition of Section XI of the ASME Code, as exempted by the methods of the analyses in Code Case N-640. The staff has determined that the proposed curves will continue to provide an acceptable level of margin and safety, and provide sufficient assurance that the LaSalle, Units 1 and 2, reactors will be operated in a manner that will protect the RPV against brittle fracture. However, given that the information specified in ComEd's submittal of June 26, 2000, and GE-NE-523-A166-1294 Report, "LaSalle Unit 1 Reactor Pressure Vessel Surveillance Materials Testing and Analysis," failed to establish the basis for the evaluation of the fluence used in the LaSalle, Units 1 and 2, P-T curve evaluation, the staff concludes that the proposed P-T limit curves are approved for use only until December 15, 2002. The proposed curves are, therefore, approved for incorporation into the LaSalle, Units 1 and 2, TSs and for use until December 15, 2002.

5.0 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.

6.0 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

(65 FR 17911). 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.

7.0 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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Date: November 8, 2000