January 16, 1990

Docket Nos. 50-373
and 50-374

Mr. Thomas J. Kovach
Nuclear Licensing Manager
Commonwealth Edison Company
Nuclear Operations
1400 Opus Place, Suite 300
Downers Grove, Illinois 60515

Distribution

Docket File	NRC & Local PDRs
PDIII-2 r/f	LLuther
JZwolinski	JCraig
OGC	DHagar
EJordan	GHill (8)
JTsao	ACRS (10)
ARM/LFMB	GPA/PA
Plant file	PShemanski
JBradfute	WJones

Dear Mr. Kovach:

SUBJECT: ISSUANCE OF AMENDMENT NO. 71 TO FACILITY OPERATING LICENSE NO. NPF-11 AND AMENDMENT NO. 55 TO FACILITY OPERATING LICENSE NO. NPF-18 - LASALLE COUNTY STATION, UNITS 1 and 2 TAC NOS. (74516/74517)

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 71 to Facility Operating License No. NPF-11 and Amendment No. 55 to Facility Operating License No. NPF-18 for the LaSalle County Station, Units 1 and 2. These amendments are in response to your letter dated June 21, 1989.

The amendments revise the LaSalle County Station, Units 1 and 2, Technical Specification 3/4 4.6 regarding pressure - temperature limits of the Reactor Coolant System in order to comply with Generic Letter 88-11 and Regulatory Guide 1.99, Revision 2.

A copy of the related Safety Evaluation supporting Amendment No. 71 to Facility Operating License No. NPF-11 and Amendment No. 55 to Facility Operating License No. NPF-18 is enclosed. Notice of issuance of these amendments will be published in the Commission's next regular biweekly Federal Register Notice.

Sincerely,
/s/
Paul C. Shemanski

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

Enclosures:
1. Amendment No. 71 to NPF-11
2. Amendment No. 55 to NPF-18
3. Safety Evaluation

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PDR ADOCK 05000373
P PDC

cc w/enclosure:
See next page
*See previous concurrence

OFC	:PDIII-2:LA	:PDIII-2:PM	:PDIII-2:PM	:PDIII-2:PD	:OGC
NAME	:LLuther	:JBradfute:ta	:PShemanski	:JCraig	:BMB
DATE	:1/9/90	:12/15/89	:12/26/89	:1/16/90	:12/29/89

OFFICIAL RECORD COPY Document Name: AMEND1 11/89

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11

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and 50-374

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Nuclear Licensing Manager
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1400 Opus Place, Suite 300
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SUBJECT: ISSUANCE OF AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. NPF-11 AND AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. NPF-18 - LASALLE COUNTY STATION, UNITS 1 and 2 TAC NOS. (74516/74517)

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The amendments revise the LaSalle County Station, Units 1 and 2, Technical Specification 3/4 4.6 regarding pressure - temperature limits of the Reactor Coolant System in order to comply with Generic Letter 88-11 and Regulatory Guide 1.99, Revision 2.

A copy of the related Safety Evaluation supporting Amendment No. to Facility Operating License No. NPF-11 and Amendment No. to Facility Operating License No. NPF-18 is enclosed.

Sincerely,

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

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

cc w/enclosure:
See next page

OFC	:PDIII-2:LA	:PDIII-2:PM	:PDIII-2:PM	:PDIII-2:PD	:OGC	:	:
NAME	:LLuther	:JBradfute:ta	:PShemanski	:JCraig	:BmB	:	: :
DATE	:11/ /89	:15/Dec/89	:26/Dec/89	:11/ /89	:12/29/89	:	:

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Document Name: AMEND1 11/89

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

Resident Inspector/LaSalle, NPS
U.S. Nuclear Regulatory Commission
Rural Route No. 1
P. O. Box 224
Marseilles, Illinois 61341

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

Attorney General
500 South 2nd Street
Springfield, Illinois 62701

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

Mr. Michael C. Parker, Chief
Division of Engineering
Illinois Department of Nuclear Safety
1035 Outer Park Drive, 5th Floor
Springfield, Illinois 62704

Regional Administrator, Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road, Bldg. #4
Glen Ellyn, Illinois 60137

John W. McCaffrey
Chief, Public Utilities Division
SOIC
100 West Randolph Street
Chicago, Illinois 60601

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-373

LASALLE COUNTY STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 71
License No. NPF-11

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

   A. The application for amendment filed by the Commonwealth Edison Company (the licensee), dated June 21, 1989 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. 71 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective upon date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John W. Craig, Director
Project Directorate III-2
Division of Reactor Projects - III, IV, V and Special Projects

Enclosure:
Changes to the Technical Specifications

Date of Issuance: January 16, 1990

ENCLOSURE TO LICENSE AMENDMENT NO. 71

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
XIX	XIX
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
B3/4 4-4	B3/4 4-4
B3/4 4-5	B3/4 4-5
B3/4 4-7	

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LIST OF FIGURES

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 Figures 3.4.6.1-1 and 3.4.6.1-1a (1) curves A for hydrostatic or leak testing; (2) curves B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curves C 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 80°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 and 3.4.6.1-1a curves A or B, as applicable, at least once per 30 minutes.

*During shutdown conditions for hydrostatic or leak testing or heatup by nonnuclear means the average coolant temperature limit of Table 2.1 for cold shutdown and hot shutdown may be increased to 212°F.

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 Figures 3.4.6.1-1 and 3.4.6.1-1a curves C 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 in accordance with the schedule in Table 4.4.6.1.3-1. The results of these fluence determinations shall be used to update the curves of Figures 3.4.6.1-1 and 3.4.6.1-1a.

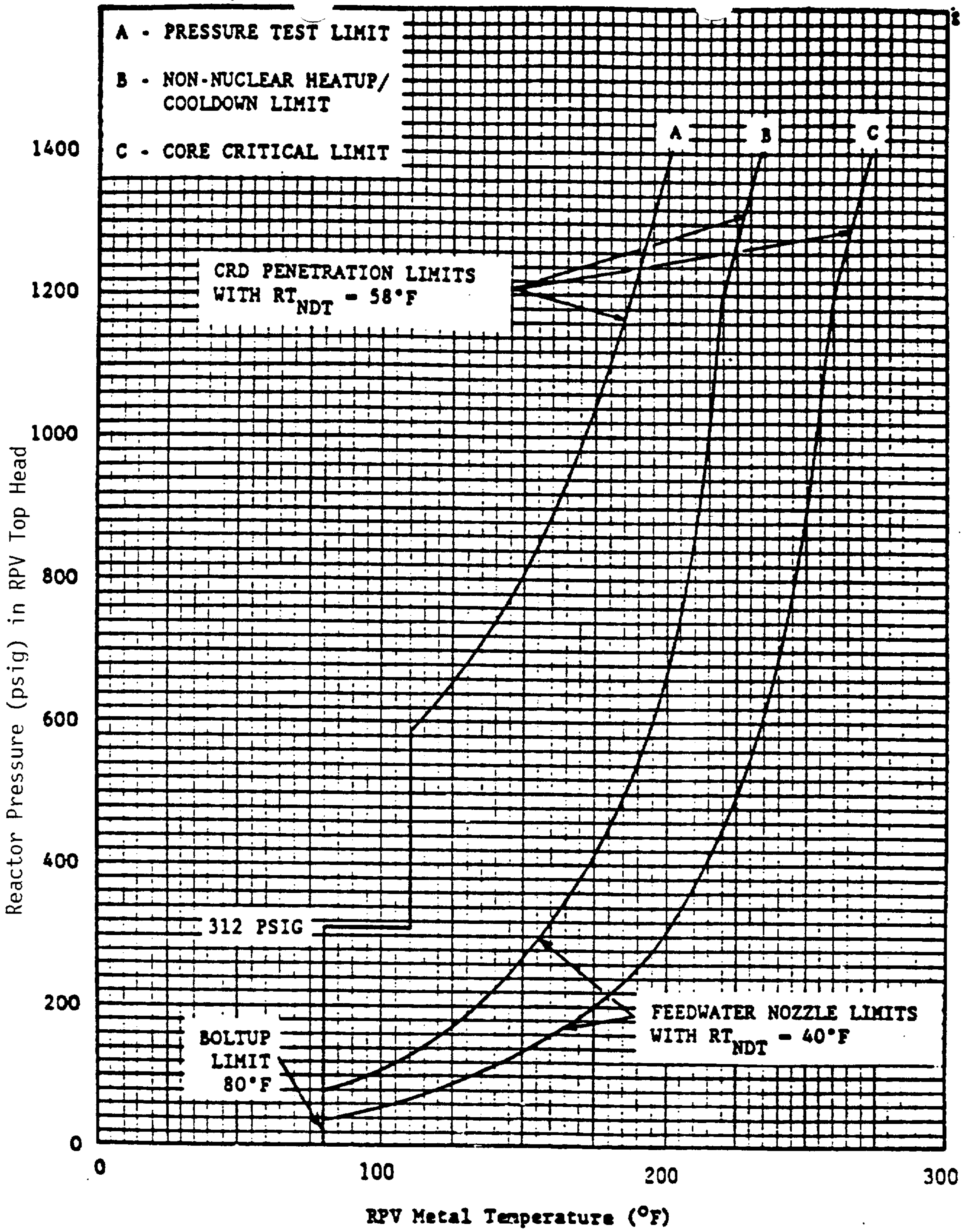
4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 80°F:

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

   1. $\leq$ 100°F, at least once per 12 hours.

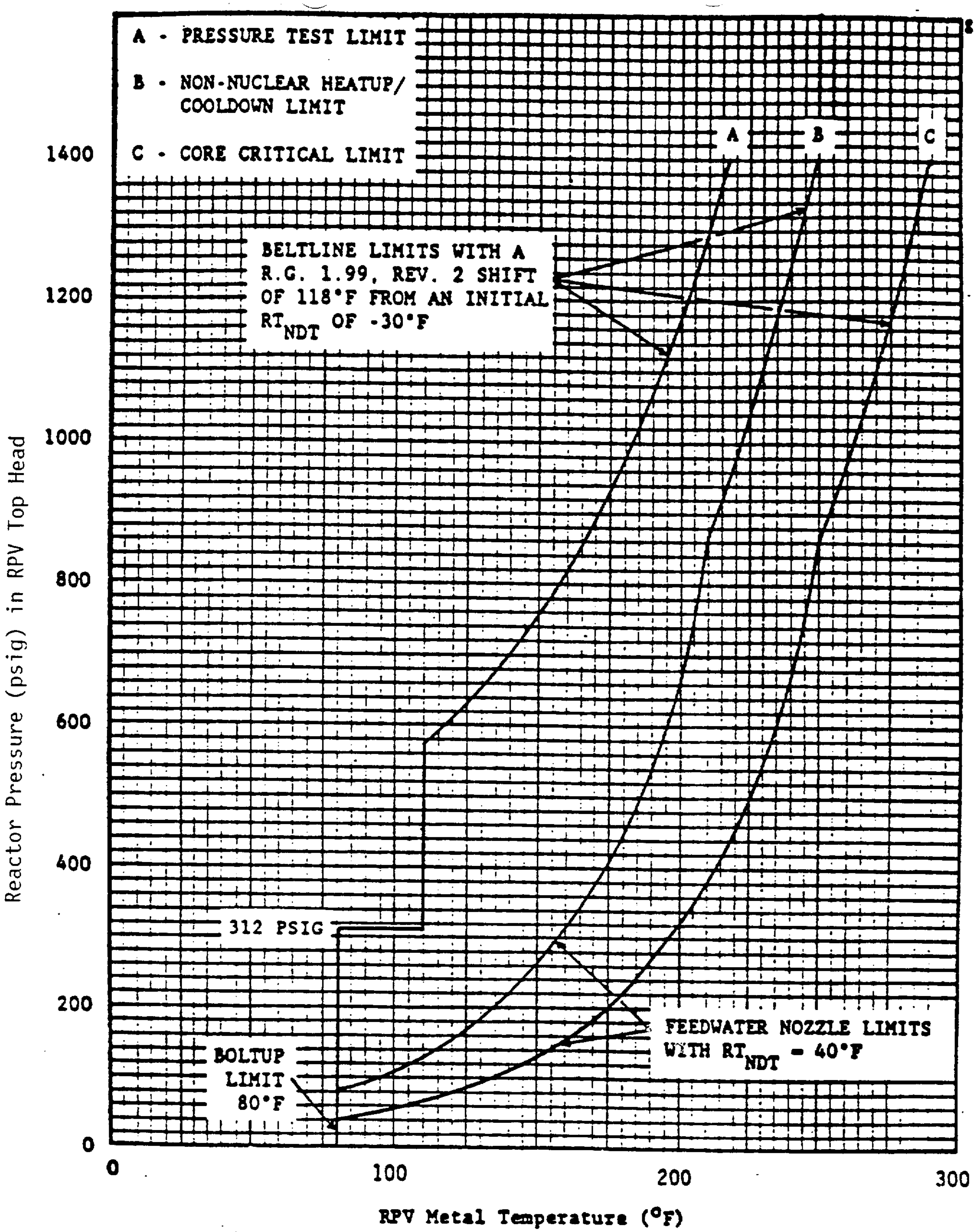
   2. $\leq$ 85°F, at least once per 30 minutes.

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



Minimum Reactor Vessel Metal Temperature vs Reactor Vessel Pressure

Figure 3.4.6.1-1



Minimum Reactor Vessel metal Temperature vs.
Reactor Vessel Pressure

Figure 3.4.6.1-1a

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 shown in Table B 3/4.4.6-1. 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, Figure 3.4.6.1-1, includes predicted adjustments for this shift in RT_{NDT} at the end of sixteen effective full power years (EFPY) while Figure 3.4.6.1-1a includes predicted adjustments in RT_{NDT} at the end of life fluence.

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. Since the neutron spectra at the material specimens and vessel inside radius are essentially identical, the irradiated specimens can be used with

PRESSURE/TEMPERATURE LIMITS (Continued)

confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figures 3.4.6.1-1 and 3.4.6.1-1a 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 and 3.4.6.1-1a for reactor criticality and for inservice leak and hydrostatic testing have been established using the requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing, General Electric "Transient Pressure Rise Affecting Fracture Toughness Requirement for Boiling Water Reactors," NEDO-21778-A, December 1978, and "Protection Against Non-Ductile Failure" of the ASME Boiler and Pressure Vessel Code, 1971 Edition, including Summer 1972 Addenda.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines 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.

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-374

LASALLE COUNTY STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 55
License No. NPF-18

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

   A. The application for amendment filed by the Commonwealth Edison Company (the licensee), dated June 21, 1989 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. 55 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective upon date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John W. Craig, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: January 16, 1990

ENCLOSURE TO LICENSE AMENDMENT NO. 55

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
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
B3/4 4-4	B3/4 4-4
B3/4 4-5	B3/4 4-5
B3/4 4-7	

LIST OF FIGURES

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 Figures 3.4.6.1-1 and 3.4.6.1-1a; (1) curves A for hydrostatic or leak testing; (2) curves B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curves C 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 80°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 and 3.4.6.1-1a curves A or B, as applicable, at least once per 30 minutes.

*During shutdown conditions for hydrostatic or leak testing or heatup by nonnuclear means, the average coolant temperature limit of Table 2.1 for cold shutdown and hot shutdown may be increased to 212°F.

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 Figures 3.4.6.1-1 and 3.4.6.1-1a curves C 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 in accordance with the schedule in Table 4.4.6.1.3-1. The results of these fluence determinations shall be used to update the curves of Figures 3.4.6.1-1 and 3.4.6.1-1a.

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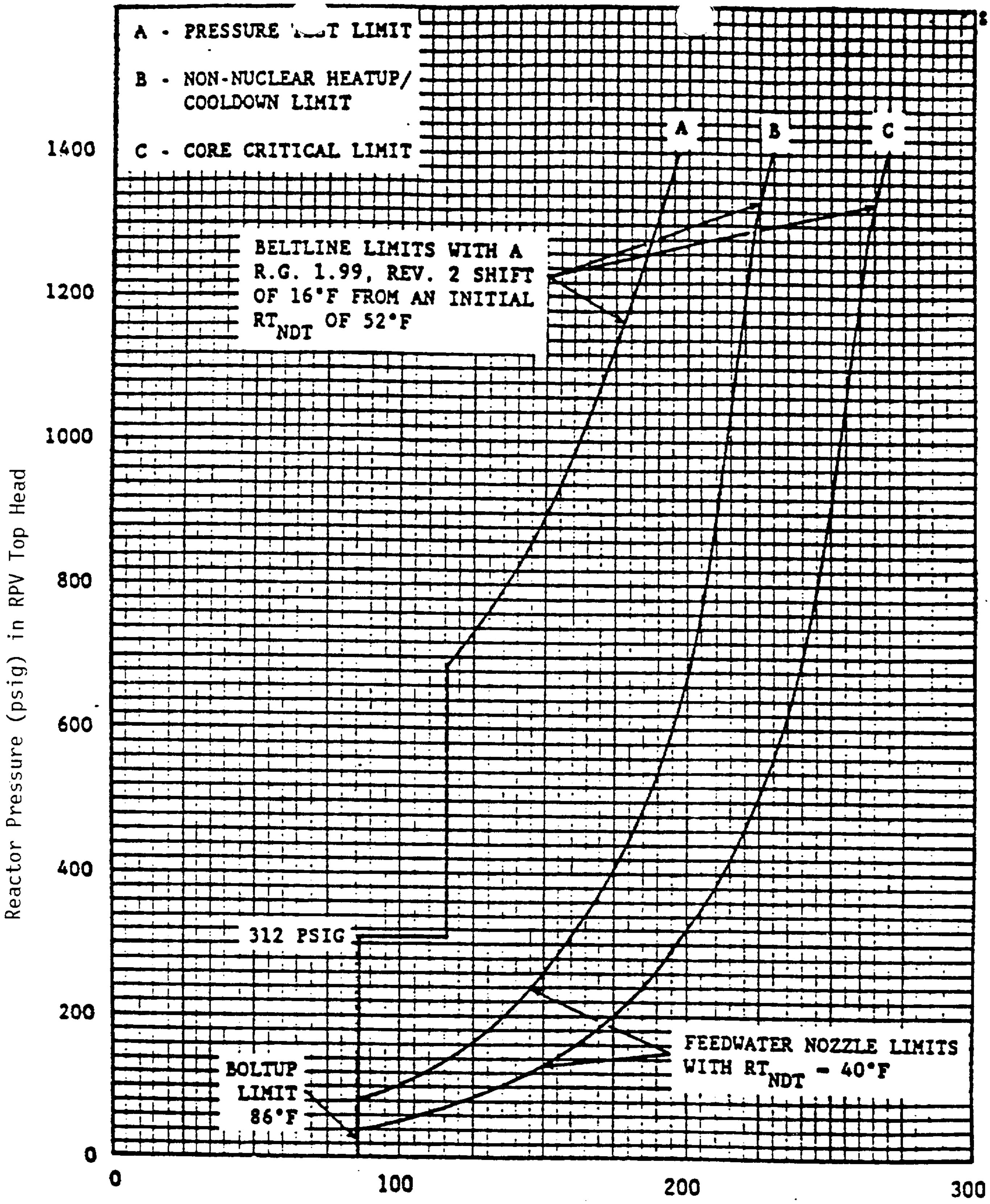
b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

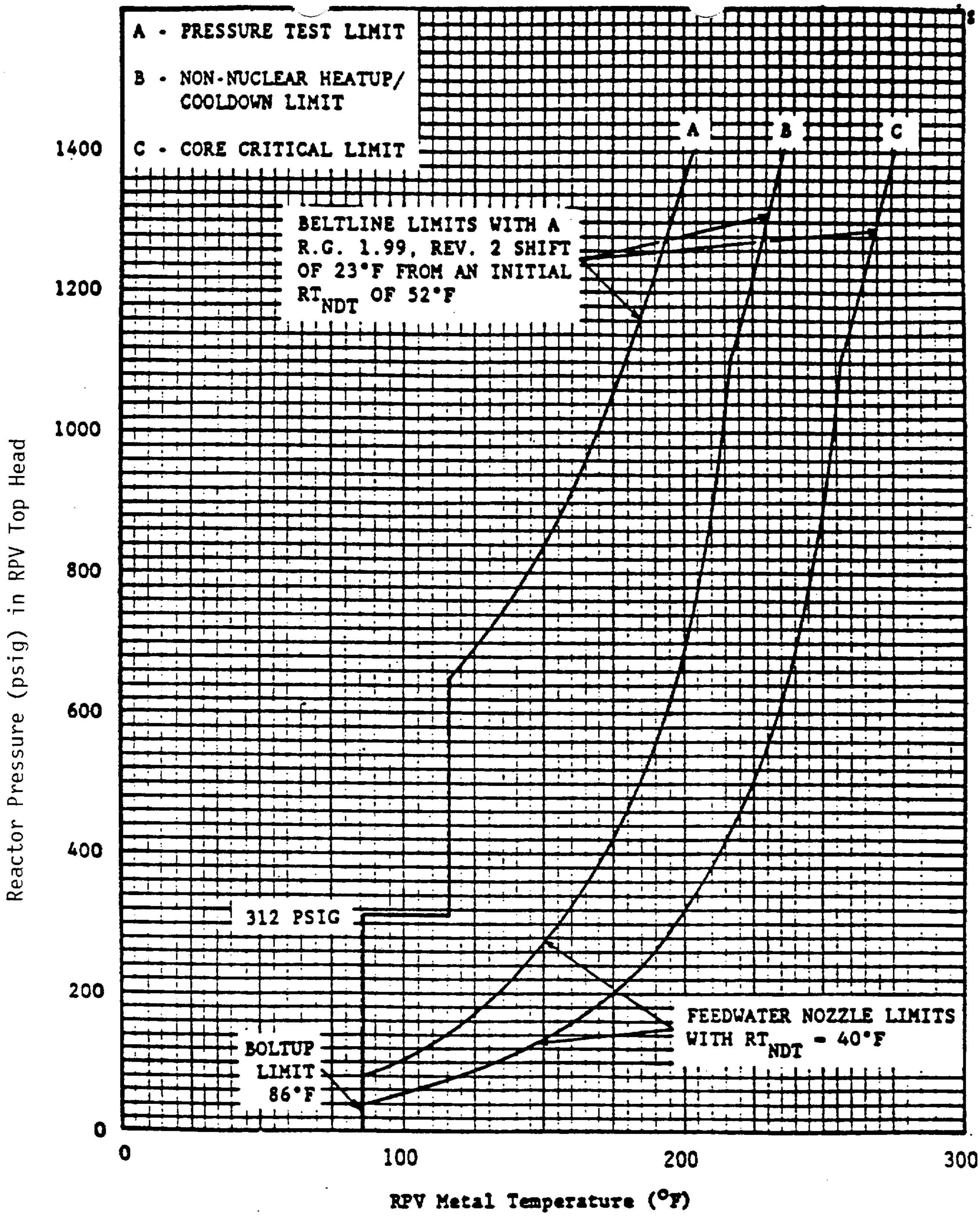
Valid to 16 EFPY



Minimum Reactor Vessel Metal Temperature vs
Reactor Vessel Pressure

Figure 3.4.6.1-1

Valid to 32 EFPY



Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure

Figure 3.4.6.1-1a

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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.

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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. Since the neutron spectra at the material specimens and vessel inside radius are essentially identical, the irradiated specimens can be used with

PRESSURE/TEMPERATURE LIMITS (Continued)

confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figures 3.4.6.1-1 and 3.4.6.1-1a 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 and 3.4.6.1-1a for reactor criticality and for inservice leak and hydrostatic testing have been established using the requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing, General Electric "Transient Pressure Rise Affecting Fracture Toughness Requirement for Boiling Water Reactors," NEDO-21778-A, December 1978, and "Protection Against Non-Ductile Failure" of the ASME Boiler and Pressure Vessel Code, 1971 Edition, including Summer 1972 Addenda.

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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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 71 TO FACILITY OPERATING LICENSE NO. NPF-11 AND

AMENDMENT NO. 55 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

In response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Effect on Plant Operations," the Commonwealth Edison Company (the licensee) requested permission to revise the pressure/temperature (P/T) limits in the LaSalle County Station Technical Specifications, Section 3.4. The request was documented in a letter from the licensee dated June 21, 1989. This revision also changes the effectiveness of the P/T limits of 16 and 32 effective full power years (EFPY). The licensee proposed to use one set of P/T limits for each unit. The proposed P/T limits were developed using Section 1 of Regulatory Guide (RG) 1.99, Revision 2. The proposed revision provides up-to-date P/T limits for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Revision 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. The P/T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear plants in the U.S. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Revision 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

2.0 EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the LaSalle 1 and 2 reactor vessels. The amount of irradiation embrittlement was calculated in accordance with Section 1 of RG 1.99, Revision 2. The staff has determined that the material with the highest ART at 16 EFPY for LaSalle 1 was lower shell plate C5978-2 with 0.11% copper (Cu), 0.59% nickel (Ni), and an initial RT_{NDT} of 23°F. The material with the highest ART at 16 EFPY for LaSalle 2 was lower-intermediate shell plate C9404-2 with 0.07% Cu, 0.49% Ni, and an initial RT_{NDT} of 52°F.

The licensee has not removed any surveillance capsules from LaSalle 1 or 2. The staff has checked the contents of all surveillance capsules and has ascertained that they contain Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material in LaSalle 1, plate C5978-2, the staff calculated the ART to be 42.3°F at 1/4T (T = reactor vessel beltline thickness) for 16 EFPY, and at 3/4T, 33.6°F. The staff used a neutron fluence of 1.3E17 n/cm^2 at 1/4T and 5.0E16 n/cm^2 at 3/4T. For the limiting beltline material in LaSalle 2, plate C9404-2, the staff calculated the ART at 16 EFPY to be 64.3°F at 1/4T for a neutron fluence of 1.4E17 n/cm^2 and 59.4°F at 3/4T for a neutron fluence of 6E16 n/cm^2.

For 32 EFPY, the limiting material in LaSalle 1 are vertical welds 3-308 A, B, and C with 0.37% Cu, 0.75% Ni, and an initial RT_{NDT} of -30°F. The staff calculated the ART at 1/4T to be 66.0°F and 25.7°F at 3/4T. The neutron fluences used were 2.6E17 and 1.0E17 n/cm^2, respectively. For LaSalle 2, the limiting material at 32 EFPY is still the lower-intermediate shell plate C9404-2 with an ART at 1/4T of 70.9° and at 3/4T of 63.9°F. The neutron fluences used were 2.8E17 and 1.2E17 n/cm^2, respectively.

The licensee used the method in Section 1 of RG 1.99, Revision 2, to calculate ART values for the inner diameter (I.D.) at 16 and 32 EFPY for LaSalle 1 and 2. The licensee's values agree with the staff's values for ART at the I.P. at 16 and 32 EFPY for both units. Substituting the ART values described above into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 40°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. At the time the LaSalle 1 and 2 reactor vessels were ordered, there were no requirements that the USE data exist. In SSER No. 2 (Reference 8), the staff reported that the licensee supplied data and analyses to demonstrate that all the beltline materials in LaSalle 1 attain an unirradiated USE of greater than 75 ft-lb. Using an EOL fluence of 3.9E17 n/cm^2 with Figure 2 of RG 1.99, Revision 2, the staff has calculated that a weld with an unirradiated USE of 75 ft-lb would have a USE of 56 ft-lb after irradiation. This is greater than 50 ft-lb and, therefore, is acceptable.

For LaSalle 2, to assure that the 50 ft-lb requirement of Section IV.B of Appendix G is met, the staff determined that the unirradiated USE required was 57 ft-lb for welds and 58 ft-lb for plate materials. In SSER No. 2 (Ref. 8), the staff reported that the licensee supplied data and analyses to demonstrate that both required unirradiated USE values have been met for all the beltline materials.

Further, it was requested that the 200°F limit for COLD SHUTDOWN be raised to 212°F for the purpose of performing hydrostatic or leak testing or heat-up by non-nuclear means. Raising this limit to 212°F is acceptable because the LaSalle Station is only slightly above sea level so the boiling point of water is approximately 212°F which assures that the reactor coolant, under only slight pressure, will not boil during these tests. While possibly creating some technical difficulty with the testing program, boiling under these circumstances has no safety significance because all the control rods will be fully inserted, thus creating no potential for criticality.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes to requirements with respect to installation or use of a facility component located within the restricted area, as defined in Part 20. The 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 these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these 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 this amendment.

4.0 CONCLUSION

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 16 and through 32 EFPY for both LaSalle 1 and 2 because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Revision 2 to calculate the ART. Hence, the proposed P/T limits may be incorporated into the LaSalle 1 and 2 Technical Specifications including the proposed modifications to the Bases.

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

5.0 REFERENCES

1. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988.

2. NUREG-0800, Standard Review Plan, Section 5.3.2, Pressure-Temperature Limits.

3. LaSalle Safety Evaluation Report Plan, Supplements 1 and 2.

4 T.A. Caine, "LaSalle County Station Units 1 and 2 Fracture Toughness Analysis per 10 CFR 50 Appendix G, SASR 88-10," General Electric, March 1988.

5. June 21, 1989, Letter from W. E. Morgan (CE) to T.E. Murley (USNRC), Subject: LaSalle County Station--Units 1 and 2, Application for Amendment to Facility Operating Licenses NPF-11 and NPF-18, Revision of Pressure-Temperature Curves.

6. LaSalle Final Safety Analysis Report, Section 5.2 and Technical Specifications.

7. NUREG-0519, Safety Evaluation Report Related to the Operation of LaSalle County Station Units 1 and 2, March 1981.

8. NUREG-0519, Supplement 2, Safety Evaluation Report Related to the Operation of LaSalle County Station Units 1 and 2, February 1982.

Principal Contributor: John C. Tsao NRR/EMTB

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