50-324

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

October 7, 1997

Mr. C. S. Hinnant, Vice President
Carolina Power & Light Company
Brunswick Steam Electric Plant
Post Office Box 10429
Southport, North Carolina 28461

SUBJECT: ISSUANCE OF AMENDMENT NO. 189 TO FACILITY OPERATING LICENSE NO. DPR-71 AND AMENDMENT NO. 220 TO FACILITY OPERATING LICENSE NO. DPR-62 CORRECTING AND UPDATING THE REACTOR PRESSURE VESSEL PRESSURE-TEMPERATURE LIMITS - BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 (TAC NOS. M97938 AND M97939)

Dear Mr. Hinnant:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 189 to Facility Operating License No. DPR-71 and Amendment No. 220 to Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant, Units 1 and 2 (BSEP1&2). The amendments consist of changes to the Technical Specifications (TS) and TS Bases in response to your application dated January 7, 1997, as supplemented on July 25, 1997, August 27, 1997, and September 15, 1997.

The amendments correct an error involving the transposition of two of the reactor pressure vessel (RPV) pressure-temperature (P-T) limits curves between the BSEP1&2 TS and update the hydrostatic pressure test limits curves for both units.

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By letter dated August 15, 1997, Carolina Power & Light Co. requested an exemption from the requirements of 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention for Lightwater Nuclear Power Reactors for Normal Operation." The exemption concerned the use of Appendix A of the 1992 Edition of the American Society of Mechanical Engineers (ASME) Code, Section XI, as an alternative to Appendix G of the 1989 Edition of the ASME Code, Section XI, for the determination of RPV P-T limits. The NRC staff has considered this request and determined that, pursuant to Section IV.A.2.b of Appendix G to 10 CFR 50, an exemption is not necessary in that use of Appendix A of the 1992 Edition of the ASME Code, Section XI, in this case provided P-T limits that are at least as conservative as those that would have been provided by use of Appendix G of the 1989 Edition of the ASME Code, Section XI.

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A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's bi-weekly Federal Register Notice.

Sincerely,

Original signed by:

David C. Trimble, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-325
and 50-324

Enclosures:
1. Amendment No. 189 to License No. DPR-71
2. Amendment No. 220 to License No. DPR-62
3. Safety Evaluation

cc w/enclosures: See next page
FILENAME - G:\BRUNSWIC\BR97938.AMD

PM:PDII-1	LA:PDII-1	OGC	PD:PDII-1	
DTrimble	EDunnington	RBachmann	JLyons	
10/6/97	10/7/97	10/6/97	10/7/97	
Yes/No	Yes/No	Yes/No	Yes/No	

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Mr. C. S. Hinnant
Carolina Power & Light Company

Brunswick Steam Electric Plant
Units 1 and 2

cc:

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AMENDMENT NO. 189 TO FACILITY OPERATING LICENSE NO. DPR-71 - BRUNSWICK, UNIT 1
AMENDMENT NO. 220 TO FACILITY OPERATING LICENSE NO. DPR-62 - BRUNSWICK, UNIT 2

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100087

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

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 189
License No. DPR-71

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

   A. The application for amendment filed by Carolina Power & Light Company (the licensee), dated January 7, 1997, as supplemented on July 25, 1997, August 27, 1997, and September 15, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and 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 rules and 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;

   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 attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. DPR-71 is hereby amended to read as follows:

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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 189, are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James E. Lyons, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 7, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 189

FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14
3/4 4-15	3/4 4-15
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 4-19	3/4 4-19
B3/4 4-4	B3/4 4-4

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 (1) Figure 3.4.6.1-1 for heatup by non-nuclear means, cooldown following a nuclear shutdown, and low power PHYSICS TESTS; (2) Figure 3.4.6.1-2 for operations with a critical core other than low power PHYSICS TESTS or when the reactor vessel is vented; and (3) Figures 3.4.6.1-3a or 3.4.6.1-3b, as applicable for inservice hydrostatic or leak testing, with:

a. A maximum heatup of 100°F in any one-hour period, except for inservice hydrostatic or leak testing at which time the maximum heatup shall not exceed 30°F in any one-hour period.

b. A maximum cooldown of 100°F in any one-hour period except for inservice hydrostatic or leak testing at which time maximum cooldown shall not exceed 30°F in any one-hour period.

c. A maximum temperature change limited to 10°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 temperatures greater than or equal to 70°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 fracture toughness properties of the reactor coolant system; determine that the system remains acceptable for continued operations, or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 The reactor coolant system temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

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 3.4.6.1-2 within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality.

4.4.6.1.3 The reactor material irradiation surveillance specimens shall be removed and examined to determine changes in material properties at the intervals shown in Table 4.4.6.1.3-1. The results of these examinations shall be used to update Figures 3.4.6.1-1, 3.4.6.1-2, 3.4.6.1-3a, and 3.4.6.1-3b, as applicable. The cumulative effective full power years shall be determined at least once per 18 months.

FIGURE 3.4.6.1-1
PRESSURE-TEMPERATURE LIMITS
REACTOR VESSEL

NORMAL OPERATION WITH CORE NOT CRITICAL

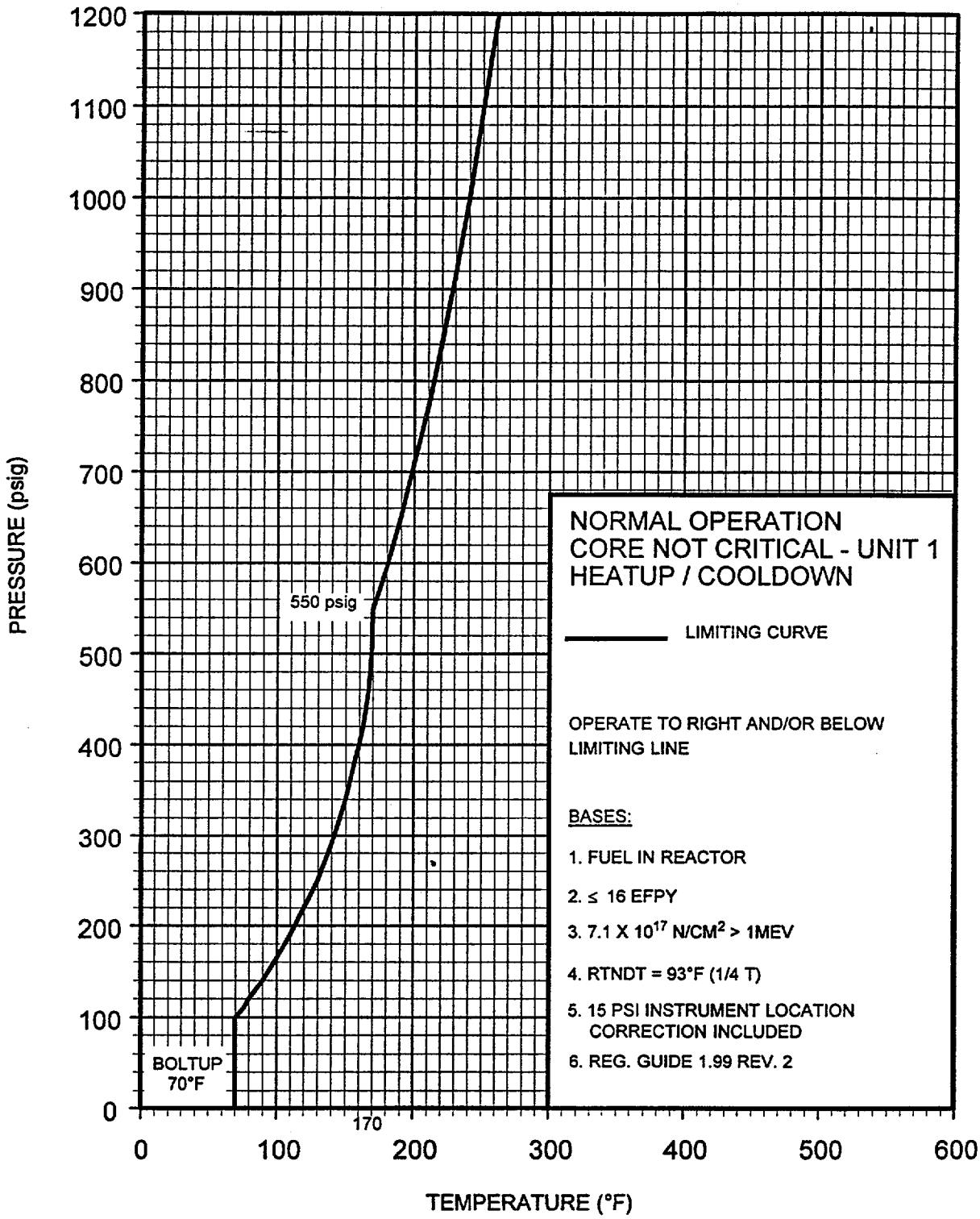


FIGURE 3.4.6.1-2
PRESSURE-TEMPERATURE LIMITS
REACTOR VESSEL

NORMAL OPERATION WITH CORE CRITICAL

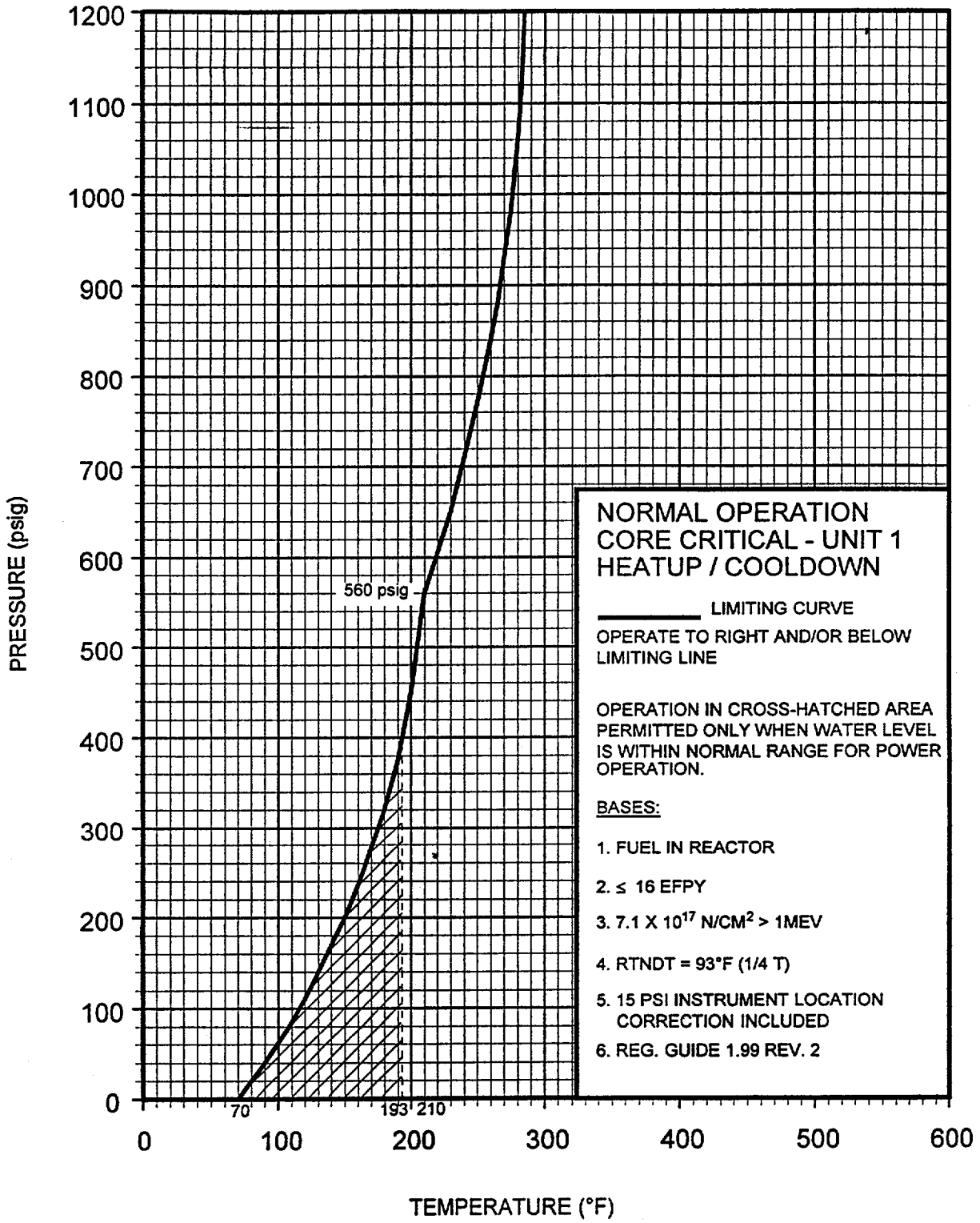


FIGURE 3.4.6.1-3a
PRESSURE-TEMPERATURE LIMITS
REACTOR VESSEL

HYDROSTATIC AND LEAK TESTS

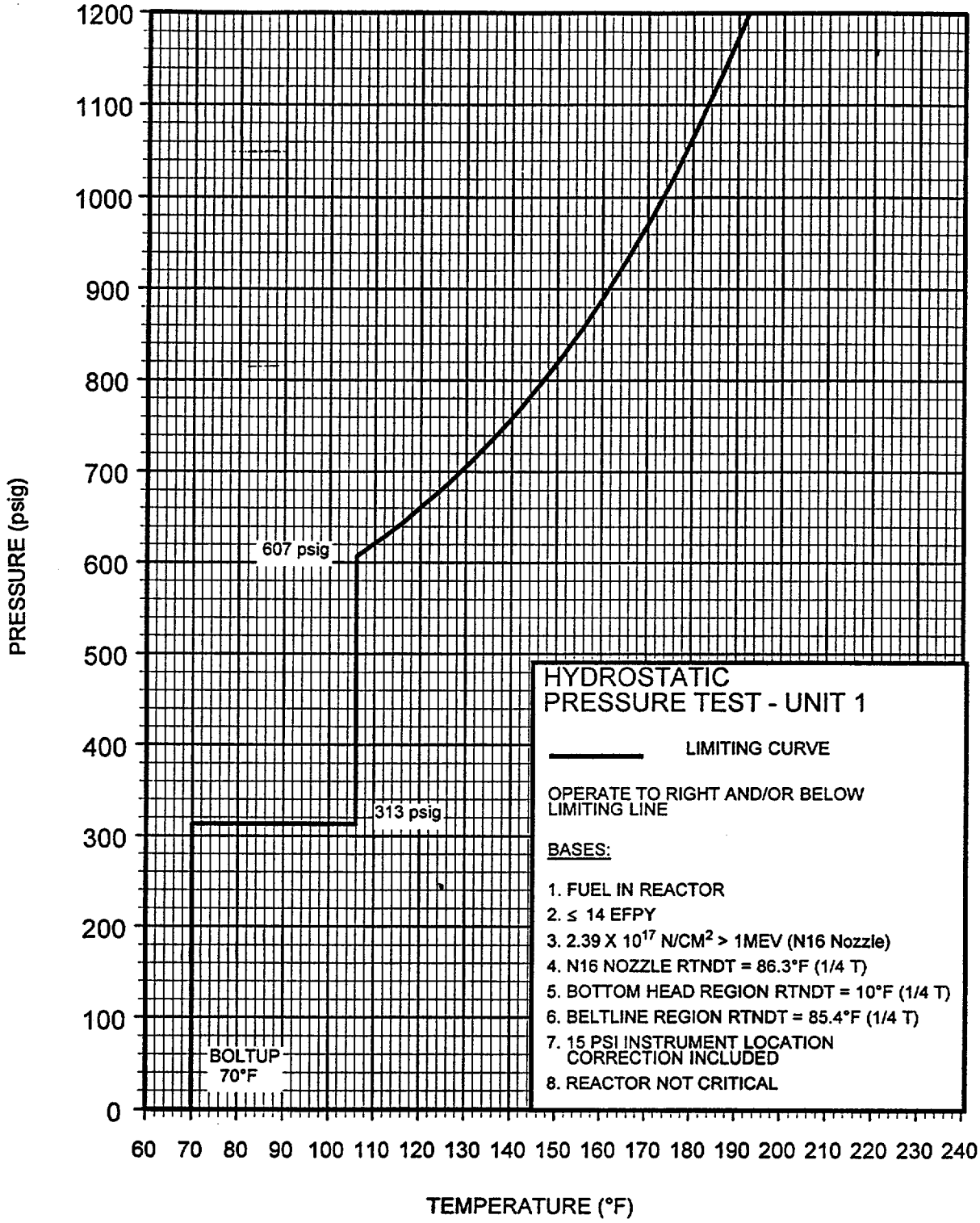
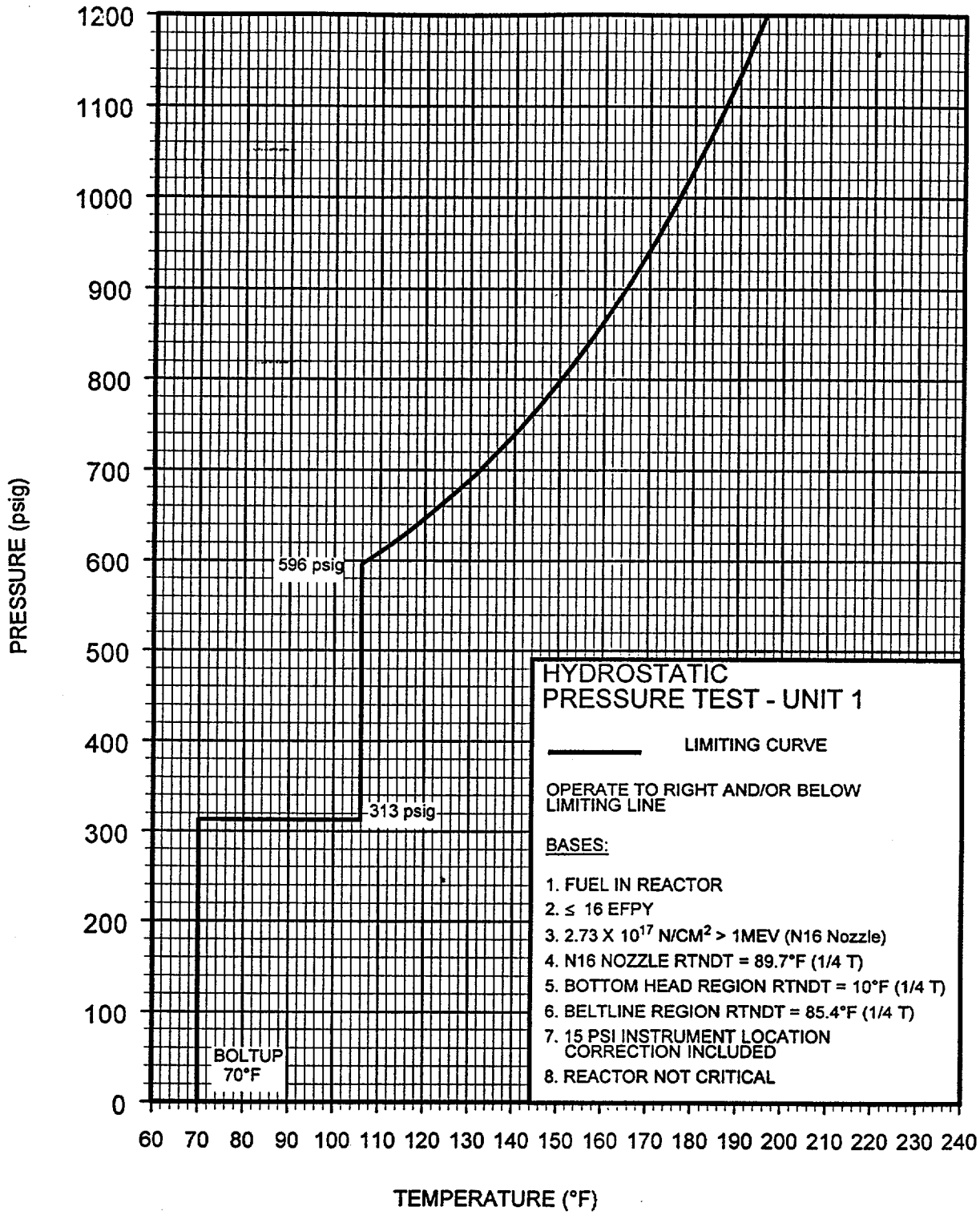


FIGURE 3.4.6.1-3b
PRESSURE-TEMPERATURE LIMITS
REACTOR VESSEL

HYDROSTATIC AND LEAK TESTS



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PRESSURE/TEMPERATURE LIMITS (Continued)

start-up 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. Thermal-induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. During cooldown, thermal gradients to be accounted for are tensile at the inner wall and compressive at the outer wall.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in GE NEDO 24161, Revision 1. Reactor operation and resultant fast neutron, E>1 Mev, fluence will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, can be predicted using the proper revision of Regulatory Guide 1.99. The pressure-temperature limit curve Figures 3.4.6.1-1, 3.4.6.1-2, 3.4.6.1-3a, and 3.4.6.1-3b include predicted adjustments for this shift in RT_{NDT} at the end of indicated EFPY, as well as adjustments to account for the location of the pressure-sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be checked periodically during operation by removing and evaluating, in accordance with ASTM E185-82, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius vary little, the measured transition shift for a sample can be adjusted with confidence to the adjacent section of the reactor vessel.

The pressure-temperature limit lines shown in Figures 3.4.6.1-1, 3.4.6.1-2, 3.4.6.1-3a, and 3.4.6.1-3b have been provided to assure compliance with the minimum temperature requirements of the 1983 revision to Appendix G of 10CFR50. The conservative method of the Standard Review Plan has been used for heatup and cooldown.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4.6.1.3-1 to assure compliance with the requirements of ASTM E185-82.

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

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 220
License No. DPR-62

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

   A. The application for amendment filed by Carolina Power & Light Company (the licensee), dated January 7, 1997, as supplemented on July 25, 1997, August 27, 1997, and September 15, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and 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 rules and 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;

   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 attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. DPR-62 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 220, are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

J. E. Lyons

James E. Lyons, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 7, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 220

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14
3/4 4-15	3/4 4-15
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 4-19	3/4 4-19
B3/4 4-4	B3/4 4-4

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 (1) Figure 3.4.6.1-1 for heatup by non-nuclear means, cooldown following a nuclear shutdown, and low power PHYSICS TESTS; (2) Figure 3.4.6.1-2 for operations with a critical core other than low power PHYSICS TESTS or when the reactor vessel is vented; and (3) Figures 3.4.6.1-3a or 3.4.6.1-3b, as applicable for inservice hydrostatic or leak testing, with:

a. A maximum heatup of 100°F in any one-hour period, except for inservice hydrostatic or leak testing at which time the maximum heatup shall not exceed 30°F in any one-hour period.

b. A maximum cooldown of 100°F in any one-hour period except for inservice hydrostatic or leak testing at which time maximum cooldown shall not exceed 30°F in any one-hour period.

c. A maximum temperature change limited to 10°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 temperatures greater than or equal to 70°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 fracture toughness properties of the reactor coolant system; determine that the system remains acceptable for continued operations, or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 The reactor coolant system temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

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 3.4.6.1-2 within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality.

4.4.6.1.3 The reactor material irradiation surveillance specimens shall be removed and examined to determine changes in material properties at the intervals shown in Table 4.4.6.1.3-1. The results of these examinations shall be used to update Figures 3.4.6.1-1, 3.4.6.1-2, 3.4.6.1-3a, and 3.4.6.1-3b, as applicable. The cumulative effective full power years shall be determined at least once per 18 months.

FIGURE 3.4.6.1-1
PRESSURE-TEMPERATURE LIMITS
REACTOR VESSEL

NORMAL OPERATION WITH CORE NOT CRITICAL

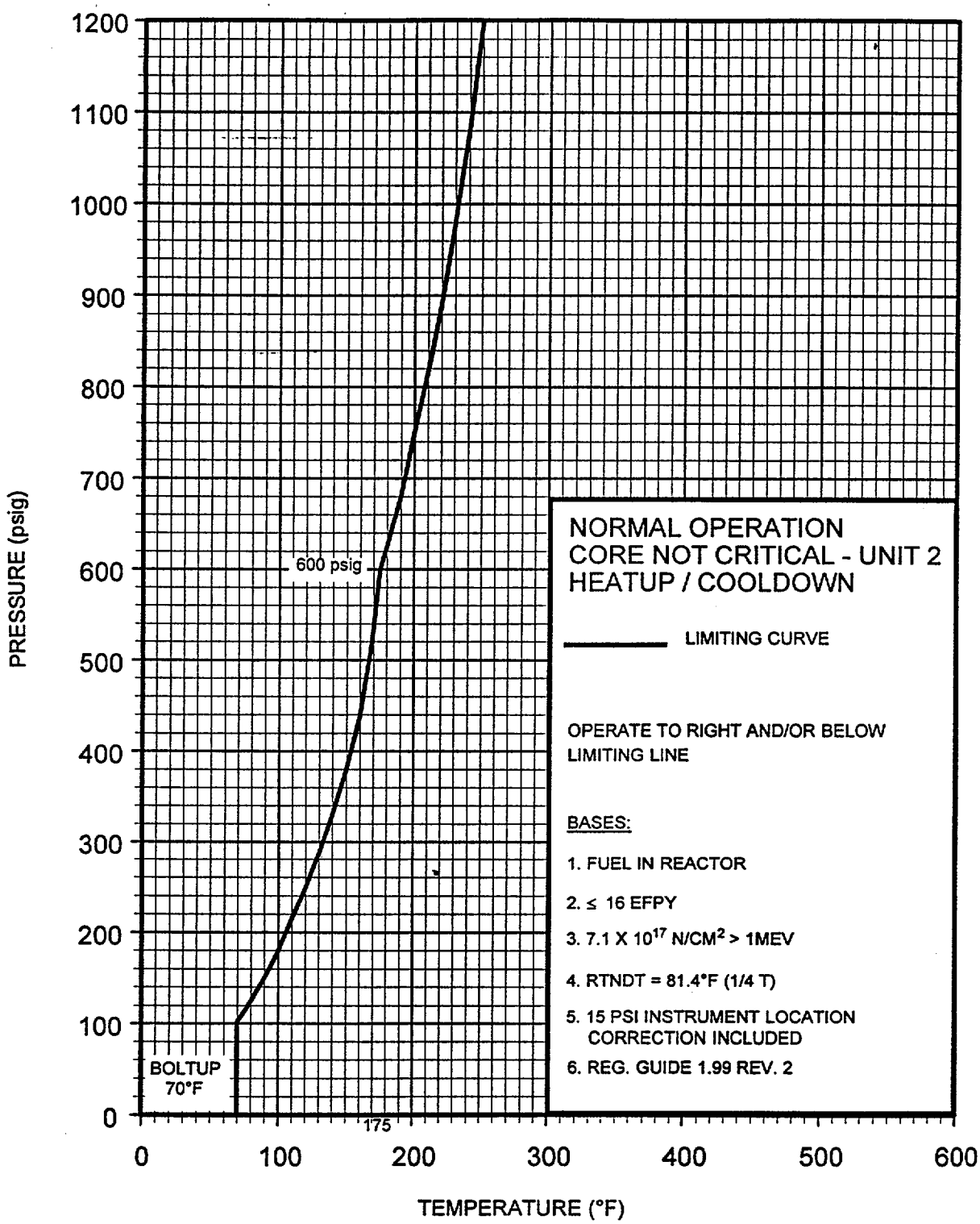


FIGURE 3.4.6.1-2
PRESSURE-TEMPERATURE LIMITS
REACTOR VESSEL

NORMAL OPERATION WITH CORE CRITICAL

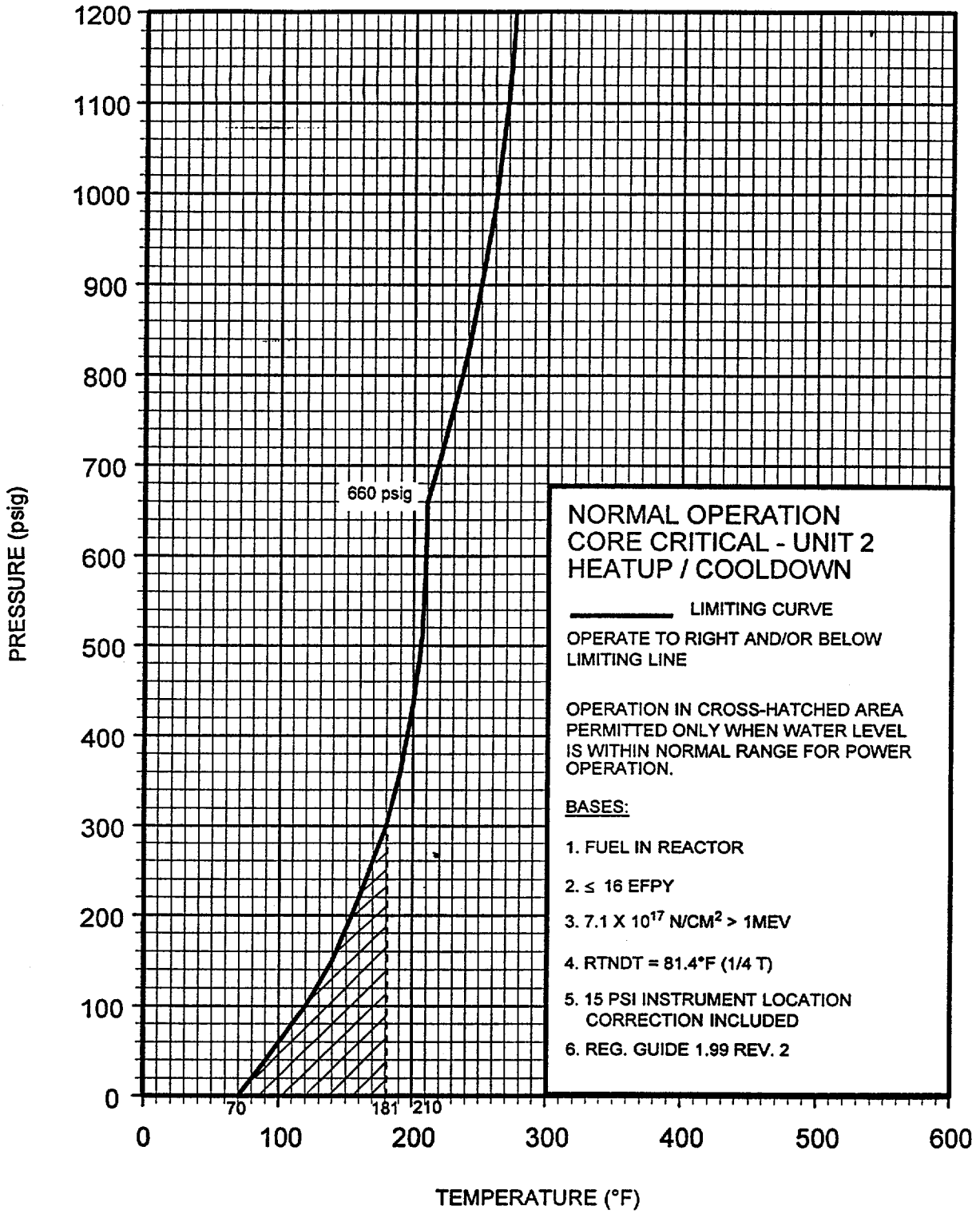


FIGURE 3.4.6.1-3a
PRESSURE-TEMPERATURE LIMITS
REACTOR VESSEL

HYDROSTATIC AND LEAK TESTS

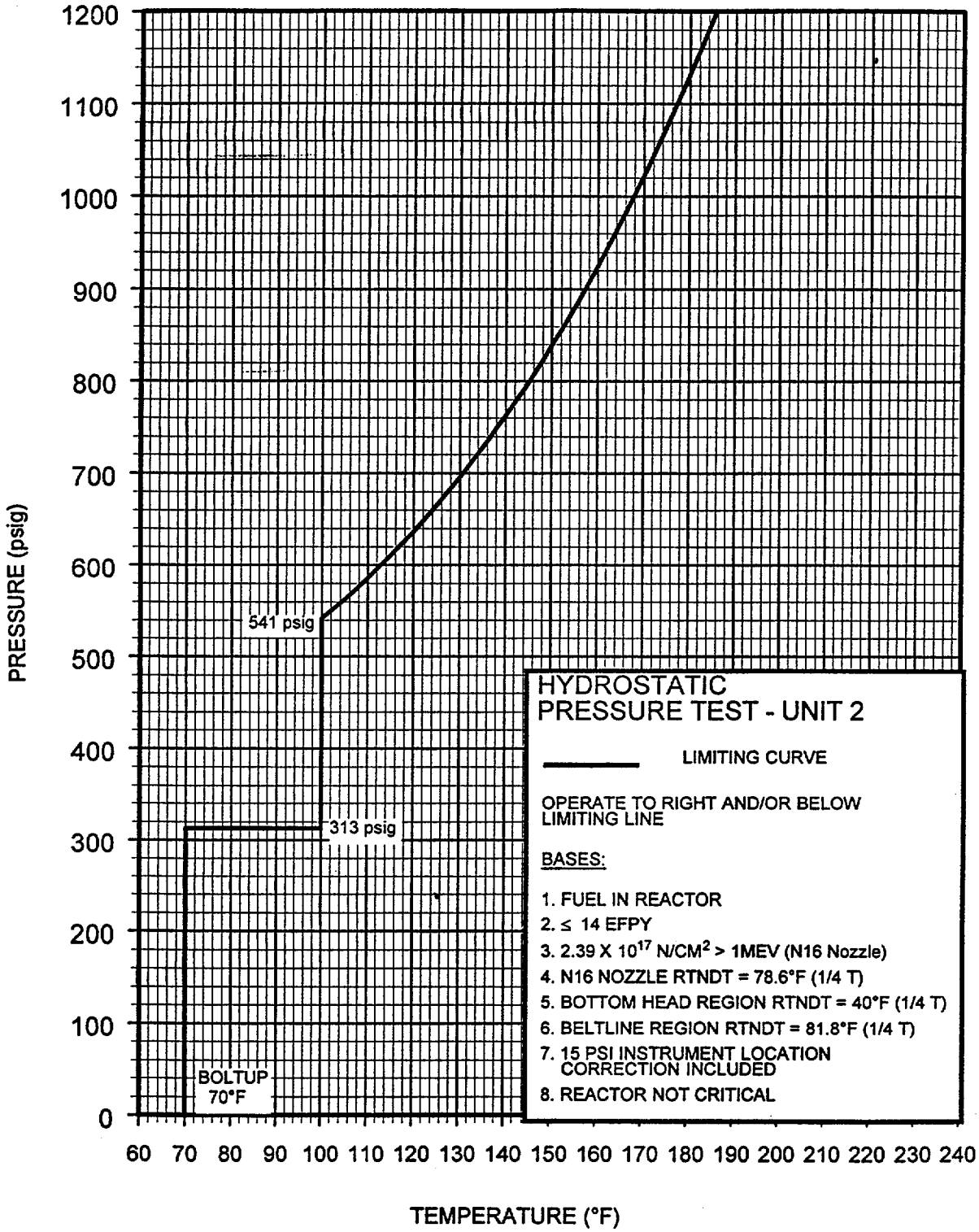
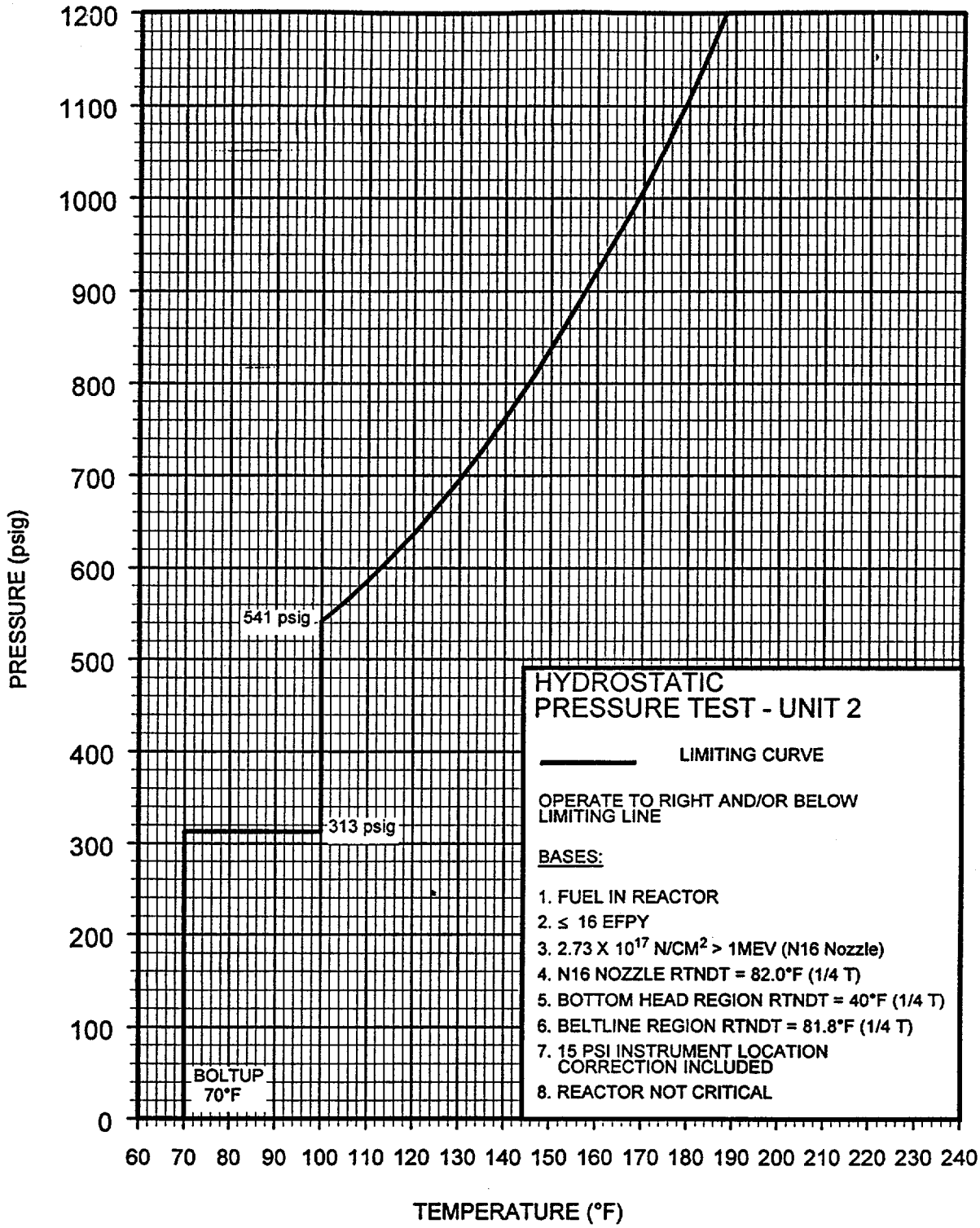


FIGURE 3.4.6.1-3b
PRESSURE-TEMPERATURE LIMITS
REACTOR VESSEL

HYDROSTATIC AND LEAK TESTS



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PRESSURE/TEMPERATURE LIMITS (Continued)

start-up 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. Thermally induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. During cooldown, thermal gradients to be accounted for are tensile at the inner wall and compressive at the outer wall.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in GE NEDO-24157, Revision 2. Reactor operation and resultant fast neutron, E>1 Mev, fluence will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, can be predicted using the proper revision of Regulatory Guide 1.99. The pressure/temperature limit curves Figures 3.4.6.1-1, 3.4.6.1-2, 3.4.6.1-3a, and 3.4.6.1-3b include predicted adjustments for this shift in RT_{NDT} at the end of indicated EFPY, as well as adjustments to account for the location of the pressure-sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be checked periodically during operation by removing and evaluating, in accordance with ASTM E185-82, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius vary little, the measured transition shift for a sample can be adjusted with confidence to the adjacent section of the reactor vessel.

The pressure/temperature limit lines shown in Figures 3.4.6.1-1, 3.4.6.1-2, 3.4.6.1-3a, and 3.4.6.1-3b have been provided to assure compliance with the minimum temperature requirements of the 1983 revision to Appendix G of 10CFR50. The conservative method of the Standard Review Plan has been used for heatup and cooldown.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4.6.1.3-1 to assure compliance with the requirements of ASTM E185-82.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 189 TO FACILITY OPERATING LICENSE NO. DPR-71

AND AMENDMENT NO.220 TO FACILITY OPERATING LICENSE NO. DPR-62

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

By letter dated January 7, 1997 (Ref. 1), the Carolina Power and Light Company (CP&L or the licensee) submitted a request to amend the Technical Specifications (TS) for the Brunswick Steam Electric Plant, Unit 1 (BSEP1) and Unit 2 (BSEP2). In the license amendment request, CP&L proposed to update the reactor pressure vessel (RPV) hydrostatic pressure test pressure-temperature (P-T) limit curves for the units to 14 and 16 effective full power years (EFPY). CP&L additionally proposed exchanging two heatup/cooldown P-T curves, TS Figures 3.4.6.1-1 (Normal Operation-Core Not Critical) and 3.4.6.1-2 (Normal Operation-Core Critical), which had been inadvertently transposed between the BSEP1 and BSEP2 TS.

A supporting calculation, Calculation Package CPL-42Q-302 (Ref. 2), performed by Structural Integrity Associates, Inc. on behalf of the licensee, was included in the amendment request. The January 7, 1997, submittal used the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, 1992 Edition, non-mandatory Appendix A (ASME Appendix A)(Ref. 3) to establish the P-T Limit Curves for the BSEP1 and BSEP2 RPVs under hydrostatic pressure test conditions. By letter dated June 20, 1997 (Ref. 4), the NRC staff provided CP&L with a Request for Additional Information (RAI). In this letter, the staff requested that CP&L provide additional information to clarify the content of Reference 1. The RAI informed CP&L that, pursuant to the requirements of 10 CFR 50.60, the staff could not process the license amendment request without submittal of an exemption request for approval to use the methodology of ASME Appendix A as an alternative to the NRC-approved methodology found in Appendix G to the 1989 Edition of the ASME Code (ASME Appendix G)(Ref. 5)[1]. CP&L provided the staff with its response to the RAI on July 25, 1997 (Ref. 6), and formally submitted the exemption request on

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1 Henceforth, Appendix G to the 1989 Edition of Section XI of the ASME Code will be referred to as ASME Appendix G. To avoid confusion, Appendix G to Part 50 of Title 10, *U.S. Code of Federal Regulations* will be referred to as 10 CFR Part 50, Appendix G. Henceforth, Appendix A to the 1992 Edition of Section XI of the ASME Code will be referred to as ASME Appendix A.

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P PDR

August 15, 1997 (Ref. 7). CP&L supplemented this information with an additional administrative change to the P-T limit curve figures in its submittal to the staff dated August 27, 1997 (Ref. 8). At the staff's request, on September 15, 1997, CP&L provided its evaluation of a January 17, 1992, cooldown event that occurred on BSEP1 during which P-T limits were exceeded (Ref. 9).

Further NRC staff review of Reference 7 and Appendix G to 10 CFR Part 50 determined that issuance of the exemption proposed by the licensee is not necessary.

The information provided in CP&L's letters of June 20, 1997, July 25, 1997, August 15, 1997, August 27, 1997, and September 15, 1997, does not affect the conclusions stated in the notice of "Proposed No Significant Hazards Consideration Determination" published in the Federal Register on March 12, 1997 (62 FR 14485).

TS 3/4.4.6 for BSEP1 and BSEP2 provides the limiting conditions for operation (LCO) and surveillance requirements (SR) for the RPVs. TS LCO 3.4.6.1 requires that the reactor coolant system (RCS) temperature and pressure be limited to the following conditions:

(a) As shown in TS Figure 3.4.6.1-1 during heat-up by non-nuclear means, cooldown following a nuclear shutdown, or low-power physics testing;

(b) As shown in TS Figure 3.4.6.1-2 for operations with a critical core other than low power physics tests or when the reactor vessel is vented; and

(c) As shown in TS Figure 3.4.6.1-3a, 3.4.6.1-3b, or 3.4.6.1-3c, as applicable, for inservice hydrostatic or leak rate testing;

In Reference 1, CP&L proposed that the following changes be approved to TS 3/4.4.6 for BSEP1 and BSEP2:

(a) Exchange TS Figures 3.4.6.1-1 and 3.4.6.1-2 between the BSEP1 and BSEP2 TS;

(b) Delete the current Figures (Figures 3.4.6.1-3a, 3.4.6.1-3b, and 3.4.6.1-3c) for the 8, 10, and 12 EFPY hydrostatic test P-T limit curves and incorporate new 14 and 16 EFPY hydrostatic test P-T limit curves for the BSEP1 and BSEP2 (proposed Figures 3.4.6.1-3a and 3.4.6.1-3b, respectively); and

(c) Reformat the TS Figures 3.4.6.1-1, 3.4.6.1-2, 3.4.6.1-3a, and 3.4.6.1-3b with administrative changes.

These changes include appropriate changes to TS LCO 3.4.6.1, SR 4.4.6.1.3, and to the TS Bases. The staff has completed its evaluation of CP&L's submittals. The staff's evaluation is provided in Section 2.0 to this Safety Evaluation (SE).

2.0 DISCUSSION AND EVALUATION

2.1 Exchange of Figure 3.4.6.1-1 of the BSEP1 TS with Figure 3.4.6.1-1 of the BSEP2 TS/Exchange of Figure 3.4.6.1-2 of the BSEP1 TS with Figure 3.4.6.1-2 of the BSEP2 TS

In Licensee Event Report (LER) 1-94-05 (Ref. 10), CP&L reported the discovery of an inadvertent transposition of the heatup-cooldown P-T limit curves for the BSEP1 RPV (Figures 3.4.6.1-1 and Figures 3.4.6.1-2) with the corresponding figures for the BSEP2 RPV. In this LER, CP&L reported that this condition had resulted in the potential use of less-conservative P-T limit curves for BSEP1 during previous heatup and cooldown evolutions. Supplemental submittals to LER 1-94-05, which were submitted to NRC on April 29, 1994 (Ref. 11), and on September 15, 1994 (Ref. 12), confirmed that the P-T limit curves were transposed and provided detailed information regarding the circumstances which led to the error. Details of the circumstances leading to the transposition of the P-T limit curves are also summarized in Reference 1.

The staff reviewed and compared BSEP1 TS Figure 3.4.6.1-1 to BSEP2 TS Figure 3.4.6.1-1, and BSEP1 TS Figure 3.4.6.1-2 to BSEP2 TS Figure 3.4.6.1-2. The BSEP1 and BSEP2 heatup/cooldown curves vary by as much as 20°F for any given reactor pressure or 100 psig for any given reactor temperature, with the P-T limits in the current curves for BSEP2 being more conservative relative to the curves for BSEP1.

10 CFR Part 50, Appendix G, requires that the P-T limits for operating nuclear plants be at least as conservative as those that would be obtained by applying the methods of ASME Appendix G. In this case, the transposition of the P-T limit curves results in the potential for heatups, cooldowns, or hydrostatic or leak rate tests of the BSEP1 reactor to be non-conservative with respect to what the actual P-T limits are for the unit. The proposed TS change will correct the non-conformance with respect to the P-T limit curves, and is therefore acceptable to the staff. It should be noted that in Reference 11 CP&L stated that a final safety evaluation would be performed to assess the safety significance of transposing the P-T limit curves after a review of all relevant plant operating data was completed.

2.2 New P-T Limit Curves Operating During Hydrostatic Testing Conditions

2.2.1 Methodologies Used for Generating the P-T Limit Data and Curves for the Beltline Materials, Bottom Vessel Head, and No. N16-A/B Instrument Nozzles

In Reference 1, CP&L provided new P-T limit curves for hydrostatic testing conditions effective to 14 EFPY and 16 EFPY (TS Figure 3.4.6.1-3a and Figure 3.4.6.1-3b, respectively). The new curves were based on an analysis that was performed by Structural Integrity Associates, Inc. (SIA), the company contracted by CP&L to generate the new curves. CP&L included SIA Calculation No. CPL-42Q-302 in Reference 1 to support its basis for establishing the new curves. SIA and the NRC staff both applied the methodology of Regulatory Guide 1.99, Revision 2 (Ref. 13), for the calculations of the Adjusted Reference Temperatures (ARTs) for the materials used to fabricate the beltline

region, bottom vessel head, and No. N16-A/B instrument nozzles. As a result of its review, the staff noted that SIA had applied a generic initial RT_{NDT} value of -56°F to the beltline welds instead of the plant-specific value of 10°F that was provided in CP&L's November 16, 1995, response to NRC Generic Letter 92-01, Revision 1, Supplement 1, "Reactor Pressure Vessel Integrity" (Ref. 14). However, the staff determined that even if the plant-specific value of 10°F were used in the calculations of the ARTs for the welds, the No. N16-A/B instrument nozzles still have the most limiting ART of all the materials in both the BSEP1 and BSEP2 RPVs. Therefore, use of a generic initial RT_{NDT} value for the weld materials did not affect the results of SIA's P-T limit assessments for the vessels.

SIA's P-T limit data for the beltline and bottom vessel head materials were based upon calculations that applied the methodology of ASME Appendix A. For evaluations of the No. N16-A/B instrument nozzles, SIA applied finite element model methods to relate the material stress intensity for the nozzles to the reactor system pressure. The staff determined that these finite element modeling methods were consistent with the finite element modeling methods of a May 1991 General Electric Company Topical Report NEDC-30634, Revision 1, "Brunswick Steam Electric Plant, Unit 1, Feedwater Nozzle Fracture Mechanics Analysis" (Ref. 15), and with the finite element methods cited in Table 6-1 of Altran Corporation Technical Report No. 96124-TR-01, Revision 0, "N-16 Nozzles Upper Shelf Energy Evaluation," December 1996 (Ref. 16), which was submitted in conjunction with CP&L's Equivalent Margins Analysis for the nozzles, dated April 14, 1997 (Ref. 17). Appendix A to this SE provides a more detailed summary of SIA's methodology for establishing the new P-T limit curves for hydrostatic testing conditions.

It should be noted that, pursuant to 10 CFR Part 50, Appendix G, the P-T limits and minimum temperatures established for RPVs must meet the requirements for these parameters set forth in Table 3 of the rule. Table 3 of the rule incorporates the P-T limit and minimum temperature requirements set forth in 1989 Edition of ASME Appendix G. Table 2.2.1-1 below summarizes the minimum temperature requirements and P-T limit requirements for hydrostatic and leak-rate testing conditions.

Table 2.2.1-1
Hydrostatic and Leak-Rate Testing Conditions

Condition	System Pressure as a Percent of the Pre-Service Hydrostatic Pressure	Requirements for P-T Limits	Minimum Temperature Requirements
Fuel in the Vessel	≤ 20%	ASME App. G Limits	Adjust. Reference Temp. (ART) of the limiting flange material
Fuel in the Vessel	> 20%	ASME App. G Limits	ART of the limiting flange material + 90°F

To meet the applicable requirements listed in Table 3 of 10 CFR Part 50, Appendix G, Paragraph 2.b. of 10 CFR Part 50, Appendix G, requires that P-T limits for operating nuclear plants must be at least as conservative as those which would be obtained by following the methods of analysis and the safety margins found in ASME Appendix G.

The staff performed an independent analysis using the methods described in ASME Appendix G in order to determine whether SIA's methods of determining the minimum allowable RCS pressures during hydrostatic testing conditions were conservative relative to the pressures which would have been obtained if the methods in ASME Appendix G were used. For the staff's evaluations of the beltline and bottom vessel head materials, the staff applied the methodology of the NRC's Standard Review Plan Chapter 5.3.2 (Ref. 18), which is consistent with the methodology found in ASME Appendix G. Appendix B to this SE provides a more detailed summary of the staff's methodology for verifying the new P-T limit curves for hydrostatic testing conditions for the beltline and bottom vessel head materials. For the staff's evaluations of the No. N16-A/B instrument nozzles, the staff used the methods described in Appendix 5 of Welding Research Council (WRC) Bulletin 175 (Ref. 19). These methods are invoked by ASME Appendix G as being acceptable methods for evaluating the inside corner of a nozzle for elastic stresses due to internal pressure. Appendix C to this SE provides a more detailed summary of the staff's methodology for verifying the P-T limit data and curves for the No. N16-A/B instrument nozzles.

2.2.2 Staff Evaluation of the Portions of the New P-T Limit Curves for Hydrostatic Testing Conditions Which Correspond to Pressures Less Than or Equal to 20% of the Preservice Hydrostatic Pressure (1563 psig)

For hydrostatic tests, Table 3 of 10 CFR Part 50, Appendix G, requires that for RCS pressures less than or equal to 20% of the preservice hydrostatic pressure, the temperature of the system must be at least as high as the ART of the limiting material in the RPV closure flange, and that the P-T limits must be at least as conservative as those which would be obtained by using the methods of ASME Appendix G. At BSEP1 and BSEP2 the preservice hydrostatic testing pressure is 1563 psig. The corresponding RCS pressure range for RCS pressures less than or equal to 20% of the preservice hydrostatic pressure is 0 psig to 313 psig. At this pressure range the minimum temperature requirement is conservative with respect to (higher than) the minimum temperatures which would be established by the P-T limit calculation requirements, and therefore is the criterion that normally establishes this portion of the curve. The ARTs of the most limiting materials in the RPV closure flange at BSEP1 and BSEP2 are 16°F and 10°F, respectively. Thus, for RCS pressures in the range of 0 psig to 313 psig, the minimum temperature of the RCS must be at least 16°F for BSEP1 and 10°F for BSEP2. CP&L has set the minimum RCS temperature for this pressure range to 70°F. This is conservative with respect to the requirements of 10 CFR Part 50, Appendix G, and therefore is acceptable to the staff.

2.2.3 Staff Evaluation of the Portions of the New P-T Limit Curves for Hydrostatic Testing Conditions Which Correspond to Pressures Greater Than 20% of the Preservice Hydrostatic Pressure (1563 psig)

For hydrostatic tests, 10 CFR Part 50, Appendix G requires that for RCS pressures greater than 20% of the preservice hydrostatic pressure (i.e, > 313 psig for BSEP1 and BSEP2), the minimum RCS temperature must be at least as high as the sum of the ART of the limiting material in the RPV closure flange and 90°F (ART + 90°F), and that the minimum temperatures established by P-T limits must be at least conservative as those which would be obtained by using the methods of ASME Appendix G. Thus to establish a hydrostatic testing P-T limit curve for pressures greater than 20% of the preservice hydrostatic pressure, the licensee must select the most conservative temperatures established by the criteria of 10 CFR Part 50, Appendix G. This is done by superimposing the minimum temperature requirement (e.g., 106°F for BSEP1 and 100°F for BSEP2) over the most limiting generated P-T limit curves for the units, and selecting the most conservative P-T data to establish the limiting composite P-T limit curve.

The staff has determined that, for the evaluation of the limiting materials in the BSEP1 beltline region and bottom vessel head, the methodology of ASME Appendix A provided P-T limit curves that were at least as conservative as those which would have resulted if the methodology of ASME Appendix G were used to establish the curves. In assessing the new hydrostatic testing P-T limit curves for BSEP-1 effective to 14 EFPY and 16 EFPY, the staff determined that the P-T limit data generated from the evaluation of the No. N16-A/B instrument nozzles were always conservative relative to the P-T limit data generated from the evaluations of the limiting materials in the reactor vessel beltline region[2] and the bottom vessel head. Thus, for BSEP1, the portions of the hydrostatic testing P-T limit curves generated for pressures greater than 313 psig are a composite generated by superimposing the minimum temperature requirement (106°F) over the curve generated from plotting the data from the evaluations of the No. N16-A/B instrument nozzles. In this case the kink in the curve for the 14 EFPY curve occurs at a temperature of 106°F (which complies with the minimum temperature requirement) and a pressure of 607 psig; at temperatures above 106°F the P-T data were generated from SIA's stress intensity calculations of the No. N16-A/B instrument nozzles. The corresponding kink for the 16 EFPY curve occurs at a temperature of 106°F and a pressure of 596 psig. The hydrostatic testing P-T limit curves generated by SIA from the evaluations of the N16-A/B instrument nozzles at BSEP-1 were slightly more conservative than the corresponding curves that were generated by the staff using the methodology of ASME Appendix G. CP&L has established

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2 It should be noted that in order to generate the data for the limiting beltline materials for BSEP1 and BSEP2 at 14 EFPY and at 16 EFPY, SIA set the ART for the beltline to the corresponding ART value of the No. N16-A/B instrument nozzles using the 16 EFPY fluence, as the forging materials for the nozzles were determined to have the highest ART's of all materials in the beltline region. The staff has determined that this is a valid engineering practice and provides an added conservatism in SIA's P-T limits evaluation of the BSEP1 and BSEP2 reactor pressure vessels.

the hydrostatic test P-T limit curves for BSEP1 (proposed TS Figures 3.4.6.1-3a and 3.4.6.1-3b effective to 14 EFPY and 16 EFPY, respectively) from the data generated by SIA from its evaluation of the No. N16-A/B instruments nozzles. Since this is the conservative practice, the staff concludes that the hydrostatic testing curves for BSEP1 effective to 14 EFPY and to 16 EFPY are acceptable for use.

For the review of the new hydrostatic testing P-T limit curves for BSEP2, the staff determined that SIA's use of the methodology of ASME Appendix A for evaluating the limiting materials in the reactor vessel beltline region and in the bottom vessel head also provided results that were at least as conservative as those which would have resulted if the methodology of ASME Appendix G were used. In assessing the new hydrostatic testing P-T limit curves for BSEP2 effective to 14 EFPY and 16 EFPY, the staff determined that the P-T limit data generated from the evaluation of the BSEP2 bottom head were conservative relative to the data generated from the evaluations of the No. N16-A/B instrument nozzles or the reactor vessel beltline region.[3] The kink in the curve for the 14 EFPY hydrostatic testing P-T limit curve for BSEP2 occurs at a temperature of 100°F (which complies with the minimum temperature requirement) and a pressure of 542 psig; at temperatures above 100°F the minimum pressures were generated from the data generated from SIA's stress intensity calculations of the bottom vessel head. For BSEP2, the hydrostatic test P-T limit curve effective to 16 EFPY is equivalent to the corresponding curve for 14 EFPY, as the stress intensities for the bottom vessel head are not dependent upon radiation effects. CP&L established the hydrostatic testing P-T limit curves for BSEP2 effective to 14 EFPY and 16 EFPY from the data generated by SIA from its evaluation of the BSEP2 bottom head. Since this is the conservative practice, the staff concludes that the hydrostatic test P-T limit curves for BSEP-2 effective to 14 EFPY and to 16 EFPY are acceptable for use.

2.3 Reformatting of TS Figures 3.4.6.1-1, 3.4.6.1-2, 3.4.6.1-3a, and 3.4.6.1-3b.

CP&L has not proposed technical changes of the current "core-not-critical" and "core-critical" heatup/cooldown curves for the BSEP1 and BSEP2 RPVs (Figures 3.4.6.1-1 and 3.4.6.1-2 in both the BSEP1 and BSEP2 TS, respectively). CP&L has proposed reformatting TS Figures 3.4.6.1-1, 3.4.6.1-2, 3.4.6.1-3a, and 3.4.6.1-3b. The formatting changes include minor changes to the axes labels and removal of the references in the curves to the limiting RCS components for which the curves were established. CP&L's proposed changes, as amended by Reference 8, constitute simple administrative changes to the existing curves, and will not affect the margins of safety established in the proposed curves nor CP&L's ability to operate BSEP1 and BSEP2 within the confines of the P-T

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3 It should be noted that the ART for No. N16-A/B instrument nozzles is the determining factor for the establishment of the BSEP1 hydrostatic P-T limit curves whereas the ART for the BSEP-2 bottom head is the determining factor for establishment of the BSEP2 hydrostatic test P-T limit curves. This difference explains how the curves were established.

limit curves during normal or transient operating conditions. These changes are therefore acceptable to the staff.

2.4 Results of Staff Review

The staff has reviewed CP&L's bases for exchanging the respective existing heatup/cooldown curves (TS Figures 3.4.6.1-1 and 3.4.6.1-2) for BSEP1 with those of BSEP2, and determined that the proposed exchange will administratively correct the error that was previously reported in Reference 10. This is acceptable to the staff.

The staff has also completed its review of the information provided by CP&L in References 1, 6, 8, and 9 in support of the TS amendment request and in Reference 7 regarding the proposed exemption. The staff has compared the licensee's new hydrostatic test P-T limit curves for BSEP1 and BSEP2, which were based on the methodology of ASME Appendix A with staff-generated curves that were based on the methodology of ASME Appendix G. The staff has determined that use of ASME Appendix A results in hydrostatic testing P-T limit curves that are at least as conservative as those that would be generated by applying the methodology of ASME Appendix G. Thus, in accordance with Section IV.A.2.b of Appendix G to 10 CFR 50, the exemption proposed in Reference 7 is not necessary. Given the staff's technical analysis, the staff concludes that, in this case, the P-T limit curves for hydrostatic testing conditions, as based on the methodology of ASME Appendix A, are therefore acceptable for use. These curves satisfy the conservatism and margin of safety requirements of 10 CFR Part 50, Appendix G, Section IV.A.2.b, regarding P-T limit curves for operating lightwater nuclear reactors, and are acceptable to the staff. The staff has also determined that the proposed administrative and corrective changes to TS 3.4.6.1 and the associated TS Bases are also acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of North Carolina official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (62 FR 11485). 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.

5.0 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.

Principal Contributor: J. Medoff

Date: October 7, 1997

REFERENCES

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2. Structural Integrity Associates, "Brunswick Units 1 and 2 Hydro Test P-T Curve Development," Calculation Package - File No. CPL-42Q-302, Revision 1, August 22, 1996.

3. American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, 1992 Edition, Section XI, Non-Mandatory Appendix A.

4. D.C. Trimble, U.S. Nuclear Regulatory Commission, letter to C.S. Hinnant, Carolina Power & Light Company, June 20, 1997.

5. American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, 1989 Edition, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure."

6. C.S. Hinnant, Carolina Power and Light Company, letter to the U.S. Nuclear Regulatory Commission, July 25, 1997.

7. C.S. Hinnant, Carolina Power and Light Company, letter to the U.S. Nuclear Regulatory Commission, August 15, 1997.

8. C.S. Hinnant, Carolina Power and Light Company, letter to the U.S. Nuclear Regulatory Commission, August 27, 1997.

9. C.S. Hinnant, Carolina Power and Light Company, letter to the U.S. Nuclear Regulatory Commission, September 15, 1997.

10. J. Cowan, Carolina Power and Light Company, letter to the U.S. Nuclear Regulatory Commission, March 22, 1994, forwarding Licensee Event Report No. 1-94-005.

11. J. Cowan, Carolina Power and Light Company, letter to the U.S. Nuclear Regulatory Commission, April 29, 1994, forwarding Supplement 1 to Licensee Event Report No. 1-94-005.

12. J. Cowan, Carolina Power and Light Company, letter to the U.S. Nuclear Regulatory Commission, September 15, 1994, forwarding Supplement 2 to Licensee Event Report No. 1-94-005.

13. U.S. Nuclear Regulatory Commission, "Radiation Embrittlement of Reactor Vessel Materials," Regulatory Guide 1.99, Revision 2, May 1988.

14. W.R. Campbell, Carolina Power and Light Company, letter to the U.S. Nuclear Regulatory Commission, November 16, 1995.

15. General Electric Company, Proprietary Topical Report NEDC-30634, Revision 1, "Brunswick Steam Electric Plant, Unit 1, Feedwater Nozzle Fracture Mechanics Analysis," May 1991.

16. Altran Corporation, Technical Report No. 96124-TR-01, Revision 0, "N-16 Nozzles Upper Shelf Energy Evaluation," December 1996.

17. W. R. Campbell, Carolina Power and Light Company, letter to the U.S. Nuclear Regulatory Commission, "Supplemental Information for Generic Letter 92-01, Reactor Vessel Structural Integrity," April 14, 1997.

18. U.S. Nuclear Regulatory Commission, Standard Review Plan, Section 5.3.2, "Pressure-Temperature Limits," NUREG-0800, Revision 1, July 1981.

19. Welding Research Council Bulletin 175, "PVRC Recommendations on Toughness Requirement for Ferritic Materials," Appendix 5, "K_I Calculation Method for the Nozzle," August 1972.

20. D.C. Trimble, U.S. Nuclear Regulatory Commission, letter to C.S. Hinnant, Carolina Power & Light Company, October , 1997.

APPENDIX A

METHODOLOGY USED BY STRUCTURAL INTEGRITY ASSOCIATES, INC.
FOR CALCULATING AND ESTABLISHING THE P-T LIMIT CURVES
FOR THE BSEP1 AND BSEP2 REACTOR VESSELS
UNDER HYDROSTATIC TESTING CONDITIONS
EFFECTIVE TO 14 AND 16 EFPY

1. P-T limit curves were generated for each of the following three reactor pressure vessel (RPV) regions: (1) beltline, which is potentially limiting due to irradiation effects on the shell forging and weld materials, (2) 2-inch instrument nozzles, which are potentially limiting due to irradiation effects and high stresses, and (3) the bottom head region, which is potentially limiting from discontinuity stresses.

2. The methodology for establishing the curves uses the following approach:

   a. Assume a temperature, T (°F).

   b. Calculate the Adjusted Reference Temperature (ART) for the limiting materials in the beltline region, the bottom vessel head, and the No. N16-A/B instrument nozzles using the methodology of Regulatory Guide 1.99, Revision 2 (1988), "Radiation Embrittlement of Reactor Vessel Materials."

   c. Calculate the critical stress intensity factor, K-IR, for the limiting materials in the beltline region, the bottom vessel head, and the No. N16-A/B instrument nozzles using the following equation:

$$K_{IR} = (1.223 * e^{[0.0145 * (T - ART + 160)]}) + 26.78 \qquad \text{(A-1)}$$

   where T is the assumed system temperature (°F), ART is the adjusted reference temperature (°F) for the material, and K_{IR} is the critical stress intensity factor (ksi$\sqrt{\text{inch}}$) for the material.

   d. Calculate the critical stress intensity, K_I, using the following equation:

$$K_I = K_{IR}/1.5 \qquad \text{(A-2)}$$

   where K_{IR} is the critical stress intensity factor (ksi$\sqrt{\text{inch}}$) for the material, and where the factor 1.5 is consistent with the safety factor required by 10 CFR Part 50, Appendix G, and ASME Appendix G for hydrostatic testing conditions.

   e. Compute the minimum allowable system pressure for the assumed reactor coolant temperature based on the following equations:

      (1) For beltline materials and the materials in the reactor vessel bottom head using

$$P = (1113.7 * K_I * t) / [F * R * ((1.0936 * t) + (0.212 * (K_I / \sigma_{ys})^2)^{0.5})] \qquad \text{(A-3)}$$

where P is the reactor pressure (psig), t is the vessel thickness (inches), F is the stress intensity proportionality factor (1.0 for the beltline materials and 1.5 for the materials in the vessel bottom head), R is the vessel inner radius, K_I is the material stress intensity factor (ksi$\sqrt{\text{inch}}$), and σ_{ys} is the vessel material yield strength (ksi).[4]

(2) For the No. 16 A/B instrument nozzles using

$$P = 21.53 * K_I \qquad \text{(A-4)}$$

where P is the reactor pressure (psig), K_I is the material stress intensity factor (ksi$\sqrt{\text{inch}}$).[5]

f. Repeat steps a. through e. using other assumed temperatures for the limiting materials in the beltline region, the bottom vessel head, and the No. N-16A/B instrument nozzles to generate P-T Limit data for the materials, and subsequently establish the P-T limit curve for hydrostatic testing using the most conservative data from the evaluations of these materials.

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4 By letter dated June 20, 1997, the NRC staff requested that CP&L provide the basis for using the pressure-stress intensity relationship equation listed above (Equation A-3) for the beltline and bottom vessel head materials. CP&L provided the basis for this equation in CP&L's letter to the staff dated July 25, 1997. The staff has determined that equation A-3 is consistent with the methodology provided in non-mandatory ASME Appendix A. CP&L submitted an exemption request to use this methodology by letter dated August 15, 1997.

5 By letter dated June 20, 1997, the NRC staff requested that CP&L provide the basis for using the pressure-stress intensity relationship equation listed above (Equation A-4) for the No. N16-A/B Instrument Nozzles. CP&L provided the basis for this equation in CP&L's letter to the staff dated July 25, 1997. The staff has determined that equation A-4 is consistent with the stress intensity equation listed in Table 6-1 of Proprietary Altran Corporation Technical Report No. 95124-TR-01, Revision 0, and with Proprietary GE Report NEDC-30634, Rev. 1, "Brunswick Steam Electric Plant, Unit 1, Feedwater Nozzle Fracture Mechanics Analysis," dated May 1991. The Altran Report was submitted with CP&L's Equivalent Margins Analysis for the No. N16-A/B Instrument Nozzles dated April 14, 1997. This methodology is based on finite element modeling of the N16-A/B Instrument Nozzles.

APPENDIX B

METHODOLOGY USED BY THE NRC STAFF
FOR VERIFYING THE P-T LIMIT DATA AND CURVES
FOR THE LIMITING MATERIALS IN THE BSEP1 AND BSEP2
BELTLINE REGIONS AND BOTTOM VESSEL HEADS
UNDER HYDROSTATIC TESTING CONDITIONS
EFFECTIVE TO 14 AND 16 EFPY

1. P-T limit curves were generated for each of the following three regions: (1) beltline, which is potentially limiting due to irradiation effects on the shell forging and weld materials, (2) 2-inch instrument nozzles, which are potentially limiting due to irradiation effects and high stresses, and (3) the bottom head region, which is potentially limiting from discontinuity stresses.

2. The methodology for verifying the P-T limit data and curves uses the following approach:

   a. Assume a temperature, T (°F).

   b. Calculate the Adjusted Reference Temperature (ART) for the limiting materials in the beltline region and the bottom vessel head using the methodology of Regulatory Guide 1.99, Revision 2 (1988), "Radiation Embrittlement of Reactor Vessel Materials."

   c. Calculate the critical stress intensity factor, K_{IR}, for the limiting materials in the beltline region and the bottom vessel head, using the following equation:

$$K_{IR} = (1.223 * e^{[0.0145 * (T - ART + 160)]}) + 26.78 \qquad \text{(B-1)}$$

   where T is the assumed system temperature (°F), ART is the adjusted reference temperature (°F) for the material, and K_{IR} is the critical stress intensity factor (ksi$\sqrt{\text{inch}}$) for the material.

   d. Calculate the critical stress intensity, K_I, using the following equation:

$$K_I = K_{IR}/1.5 \qquad \text{(B-2)}$$

   where K_{IR} is the critical stress intensity factor (ksi$\sqrt{\text{inch}}$) for the material, and where the factor 1.5 is consistent with the safety factor required by 10 CFR Part 50, Appendix G, and ASME Appendix G for hydrostatic testing conditions.

   e. Calculate the membrane stress using the following equation:

$$K_I = G * \sigma_m * M_m \qquad \text{(B-3)}$$

   where K_I is the material stress intensity factor, G represents a proportionality factor to account for increases in the stress intensity

factor due to geometric constraints[6], σ_m is the membrane stress (ksi) for the material in beltline region or bottom vessel head, and M_m is a proportionality factor defined in the Paragraph G-2214 of the 1989 Edition of the ASME Code, Section XI, Appendix G.[7]

f. Calculate the minimum allowable pressure using one of the following membrane stress equations:

(1) for evaluations of the beltline region of the reactor pressure vessels, using:

$$P = 1000 * (\sigma_m * t) / R \qquad \text{(B-4)}$$

where P is the calculated reactor pressure (psi), σ_m is the membrane stress (ksi) for the material in beltline region or bottom vessel head, t is the vessel shell thickness (inches), and r is the reactor vessel inner radius (inches),

(2) for evaluations of the bottom vessel heads, using:

$$P = 1000 * (\sigma_m * t) / (2 * R) \qquad \text{(B-5)}$$

g. Repeat steps a. through f. using other assumed temperatures to generate the P-T limit data for the beltline region and the bottom vessel head, and subsequently establish the composite P-T limit curve for hydrostatic testing using the most conservative data from the evaluations of the limit materials in the beltline region, bottom vessel head, and No. N16-A/B instrument nozzles (Note: the methodology for the staff's analysis of the nozzles is summarized in Appendix C).

h. Compare the NRC generated composite hydrostatic test P-T limit curve with the corresponding P-T limit curve presented in the proposed Technical Specifications.

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6 For plates and weld materials in the beltline region G was set equal to 1.0; for the materials in the bottom vessel head, G was set to a factor of 3.0 to account for increases in the stress intensity factor due to the presence of control rod drive mechanism (CRDM) penetration nozzles or other instrument nozzles. These factors are consistent with the factors used in the analysis by SIA, and with the geometric factors recommended in WRC Bulletin 175-1972.

7 According the Paragraph G-2214.1 of the 1989 Edition of the ASME Code, Section XI, Appendix G, "The K_i corresponding to membrane tension for the postulated defect . . . is $K_{im} = M_m$ * membrane stress, where M_m is as shown in Figure G-2214-1."

APPENDIX C

METHODOLOGY USED BY THE NRC STAFF
FOR VERIFYING THE P-T LIMIT DATA AND CURVES
FOR THE LIMITING MATERIALS IN THE BSEP1 AND BSEP2
NO. N16-A/B INSTRUMENT NOZZLES
UNDER HYDROSTATIC TESTING CONDITIONS
EFFECTIVE TO 14 AND 16 EFPY

1. P-T limit curves were generated for each of the following three regions: (1) beltline, which is potentially limiting due to irradiation effects on the shell forging and weld materials, (2) 2-inch instrument nozzles, which are potentially limiting due to irradiation effects and high stresses, and (3) the bottom head region, which is potentially limiting from discontinuity stresses.

2. The methodology for verifying the P-T limit data and curves for the No. N16-A/B instrument nozzles uses the following approach:

   a. Assume a temperature, T (°F).

   b. Calculate the Adjusted Reference Temperature (ART) for the limiting materials in the No. N16-A/B instrument nozzles using the methodology of Regulatory Guide 1.99, Revision 2 (1988), "Radiation Embrittlement of Reactor Vessel Materials."

   c. Calculate the critical stress intensity factor, K_{IR}, for the No. N16-A/B instrument nozzles using the following equation:

$$K_{IR} = (1.223 * e^{[0.0145 * (T - ART + 160)]}) + 26.78 \qquad \text{(C-1)}$$

   where T is the assumed system temperature (°F), ART is the adjusted reference temperature (°F) for the material, and K_{IR} is the critical stress intensity factor (ksi$\sqrt{\text{inch}}$) for the material.

   d. Calculate the critical stress intensity, K_I, using the following equation:

$$K_I = K_{IR}/1.5 \qquad \text{(C-2)}$$

   where K_{IR} is the critical stress intensity factor (ksi$\sqrt{\text{inch}}$) for the material, and where the factor 1.5 is consistent with the safety factor required by 10 CFR Part 50, Appendix G, and ASME Appendix G for hydrostatic testing conditions.

   e. Calculate the membrane stress using the methodology of Appendix 5 to WRC Bulletin No. 175-1972 and the following equation:

$$\sigma_m = K_I \ / \ (F(a/r_n) * (\pi * a)^{0.5}) \qquad \text{(C-3)}$$

   where K_I is the material stress intensity factor, $F(a/r_n)$ represents a proportionality factor to account for increases in the stress intensity

factor due to geometric constraints[8], the parameter "a" represents the postulated flaw size (inches) used in the fracture mechanics analysis[9], and σ_m is the membrane stress (ksi) for the material in beltline region or bottom vessel head.

f. Calculate the minimum allowable pressure using the following membrane stress equations:

$$P = 1000 * (\sigma_m * t) / R \qquad \text{(C-4)}$$

where P is the calculated reactor pressure (psi), σ_m is the membrane stress (ksi) for the material in beltline region or bottom vessel head, t is the vessel shell thickness (inches), and r is the reactor vessel inner radius (inches).

g. Repeat steps a. through f. using other assumed temperatures to generate P-T limit data for the nozzles, and subsequently establish the composite P-T limit curve for hydrostatic testing using the most conservative data from the evaluations of the limit materials in the beltline region of the reactor, the bottom vessel head, and the No. N16-A/B instrument nozzles (Note: the methodology for the staff's analysis of the beltline region and the bottom vessel head is summarized in Appendix B).

h. Compare the NRC generated P-T limit curve with the corresponding P-T limit curve presented in the proposed amended Technical Specifications.

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8 Appendix 5 of WRC Bulletin No. 175-1972 provides an acceptable method of evaluating the stress intensities in nozzles joined to reactor pressure vessel shells. Figure A5-1 of Appendix 5 to the Bulletin may be used to determine the proportionality factor $F(a/r_n)$ when used in a manner that is consistent with the Appendix's overall recommended methodology.

9 According to WRC Bulletin No. 175-1972, for evaluations of a nozzle joined to the beltline region of a reactor pressure vessel, the postulated flaw is a semi-elliptical surface flaw which initiates from the inner surface of the juncture between the nozzle and the shell cylinder. The depth of this flaw is postulated to extend one-quarter of the thickness of the nozzle along a line oriented at a 45° angle to the inner surface of the reactor pressure vessel shell. According to WRC Bulletin No. 175-1972, this juncture is a critical location because the local membrane stress at this point produced by pressure may be 2 to 3 times the magnitude of the membrane stress in other locations of the shell.