November 28, 1995

Mr. Leon R. Eliason
Chief Nuclear Officer & President-
Nuclear Business Unit
Public Service Electric & Gas
Company
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION (TAC NO. M93054)

Dear Mr. Eliason:

The Commission has issued the enclosed Amendment No. 88 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated July 27, 1995.

The Technical Specification (TS) revision incorporates updated pressure vs. temperature operating limit curves contained in TS Figure 3.4.6.1-1. The revision also changes TS Surveillance Requirement 4.4.6.1.3 based upon implementation of Regulatory Guide 1.99, Rev. 2, in accordance with Generic Letter 88-11, for reactor vessel material surveillance specimens. The changes are a result of data obtained from the first set of specimen capsules removed during refueling outage 5.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly <u>Federal Register</u> notice.

Sincerely,
/S/
David H. Jaffe, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures: 1. Amendment No. 88 to License No. NPF-57
2. Safety Evaluation

cc w/encls: See next page

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

November 28, 1995

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Mr. Leon R. Eliason
Public Service Electric & Gas
Company

Hope Creek Generating Station

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 88
License No. NPF-57

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

   A. The application for amendment filed by the Public Service Electric & Gas Company (PSE&G) dated July 27, 1995, 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 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 attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-57 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. 88 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSE&G shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance, and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: November 28, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 88

FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
3/4 4-21	3/4 4-21
3/4 4-22	3/4 4-22
3/4 4-23	3/4 4-23
---	3/4 4-23a
---	3/4 4-23b
B 3/4 4-5	B 3/4 4-5
B 3/4 4-6	B 3/4 4-6
B 3/4 4-7	B 3/4 4-7

REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 (hydrostatic or leak testing), and Figure 3.4.6.1-2 (heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS), and Figure 3.4.6.1-3 (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 metal temperature shall be maintained greater than or equal to 79°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3 as applicable, at least once per 30 minutes.

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 3.4.6.1-3 within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

4.4.6.1.3 The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties, as required by 10 CFR 50, Appendix H. The results of these examinations shall be used to update the curves of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3.

4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 70°F:

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

   1. ≤ 100°F, at least once per 12 hours.

   2. ≤ 80°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.

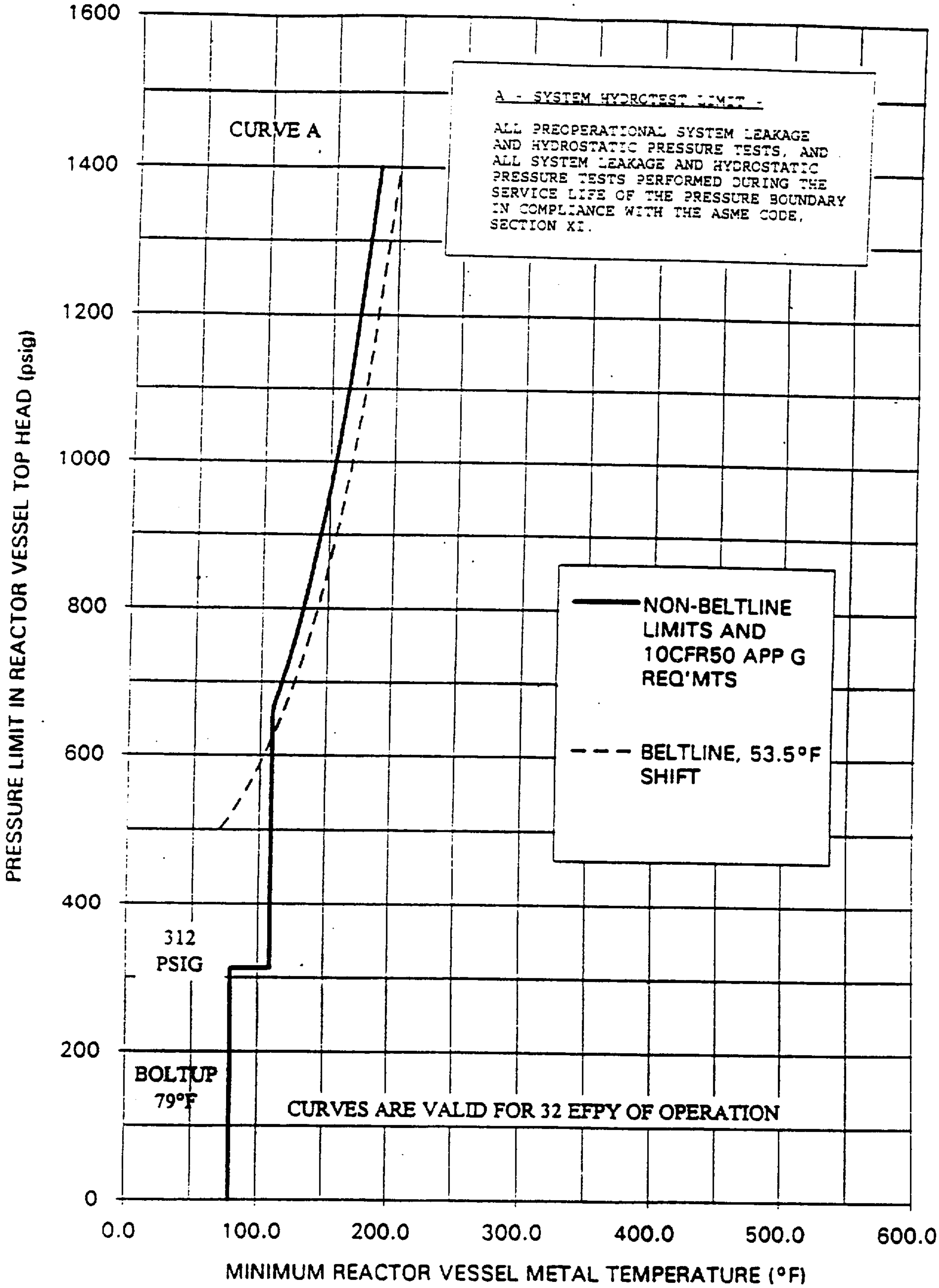
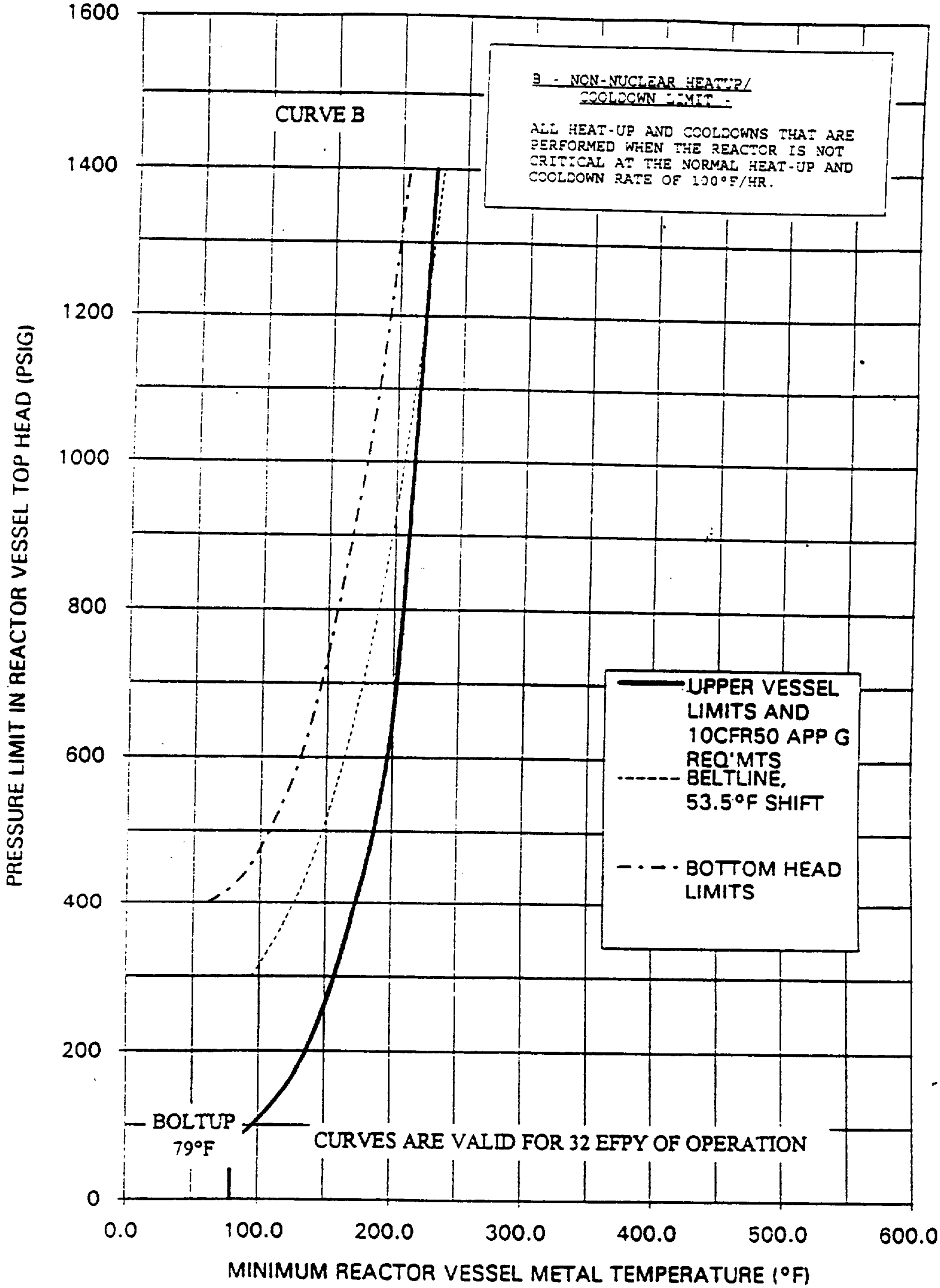
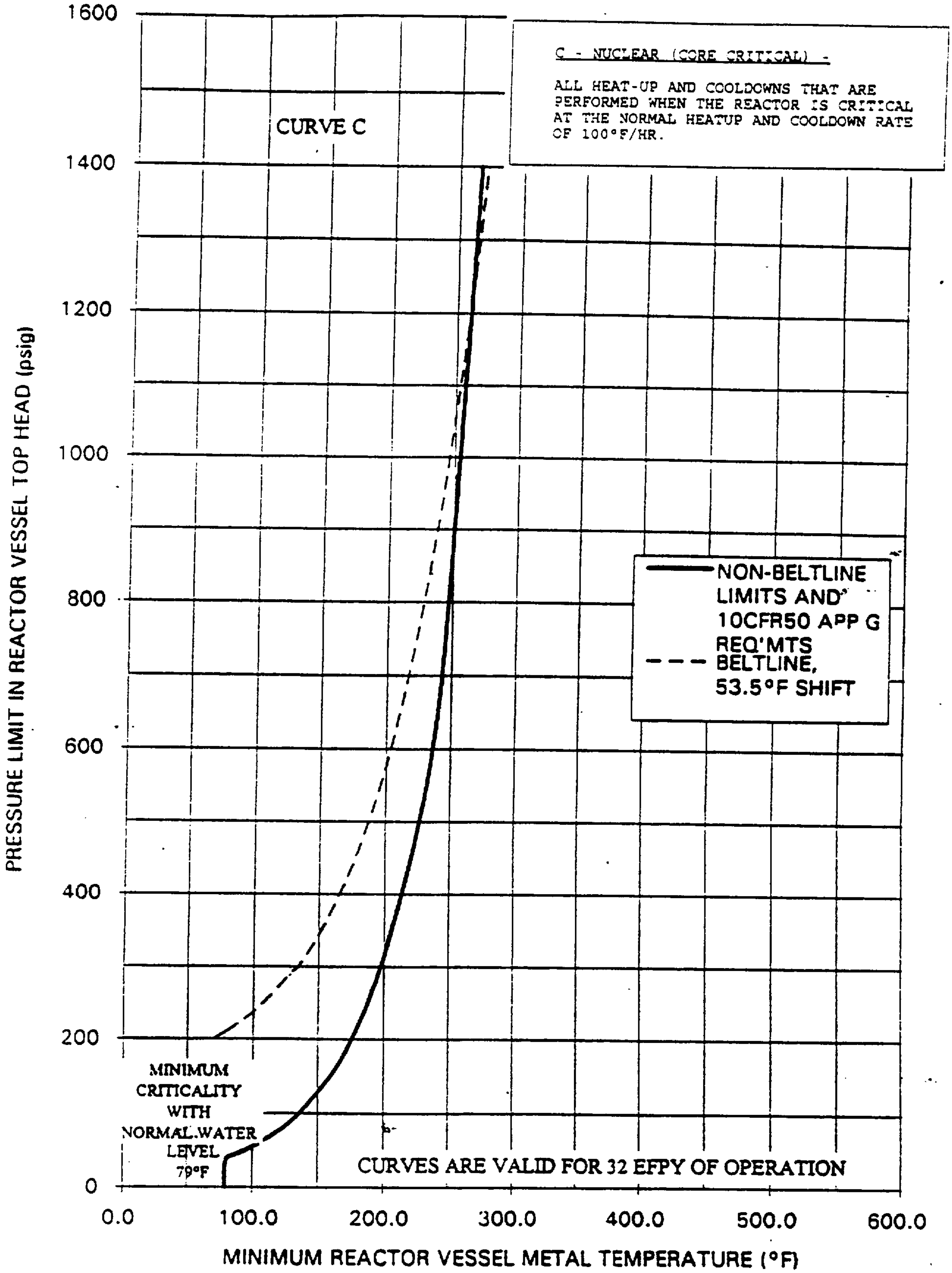


FIGURE 3.4.6.1-1



CURVE B
B - NON-NUCLEAR HEATUP/ COOLDOWN LIMIT -
ALL HEAT-UP AND COOLDOWNS THAT ARE PERFORMED WHEN THE REACTOR IS NOT CRITICAL AT THE NORMAL HEAT-UP AND COOLDOWN RATE OF 100°F/HR.
UPPER VESSEL LIMITS AND 10CFR50 APP G REQ'MTS
BELTLINE, 53.5°F SHIFT
BOTTOM HEAD LIMITS
BOLTUP 79°F
CURVES ARE VALID FOR 32 EFPY OF OPERATION
PRESSURE LIMIT IN REACTOR VESSEL TOP HEAD (PSIG)
MINIMUM REACTOR VESSEL METAL TEMPERATURE (°F)
0
200
400
600
800
1000
1200
1400
1600
0.0
100.0
200.0
300.0
400.0
500.0
600.0


CURVE C
C - NUCLEAR (CORE CRITICAL) -
ALL HEAT-UP AND COOLDOWNS THAT ARE PERFORMED WHEN THE REACTOR IS CRITICAL AT THE NORMAL HEATUP AND COOLDOWN RATE OF 100°F/HR.
NON-BELTLINE LIMITS AND 10CFR50 APP G REQ'MTS
BELTLINE, 53.5°F SHIFT
MINIMUM CRITICALITY WITH NORMAL WATER LEVEL 79°F
CURVES ARE VALID FOR 32 EFPY OF OPERATION
PRESSURE LIMIT IN REACTOR VESSEL TOP HEAD (psig)
1600
1400
1200
1000
800
600
400
200
0
0.0
100.0
200.0
300.0
400.0
500.0
600.0
MINIMUM REACTOR VESSEL METAL TEMPERATURE (°F)

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 UFSAR. 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.

The operating limit curves of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3 are derived from the fracture toughness requirements of 10 CFR 50 Appendix G and ASME Code Section III, Appendix G. The curves are based on the RT_{NDT} and stress intensity factor information for the reactor vessel components. Fracture toughness limits and the basis for compliance are more fully discussed in UFSAR Chapter 5, Paragraph 5.3.1.5, "Fracture Toughness."

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of some 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 Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Material". The pressure/ temperature limit curves, Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3, includes an assumed shift in RT_{NDT} for 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, irradiated flux wires installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the flux wires and vessel inside radius are essentially identical, the irradiated flux wires can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3 shall be adjusted, as required, on the basis of the flux wire data and recommendations of Regulatory Guide 1.99, Rev. 2.

PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown in Figures 3.4.6.1-1 and 3.4.6.1-3, curves for inservice leak and hydrostatic testing and reactor criticality have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance capsules and the frequencies for removing and testing the specimens in these capsules are provided in UFSAR Section 5.3 and Appendix 5A.

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, however, single failure considerations require that two valves be OPERABLE. 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. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

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 1977 Edition and Addenda through Summer 1978.

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.

BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

BELTLINE COMPONENT	WELD SEAM I.D. OR MAT'L TYPE	HEAT/SLAB OR HEAT/LOT	CU(%)	Ni(%)	HIGHEST RT_{NDT}(°F)	ΔRT_{NDT}(°F)	PREDICTED EOL UPPER SHELF (FT-LBS)	MAX. EOL RT_{NDT}(°F)
Plate	SA-533 GR B CL.1	5K3025-1	.15	0.71	+19	53.5	67	72.5
Weld	Vert. seams for shells 4&5	D53040/ 1125-02205	.08	0.59	-30	62.8	120	32.8

NOTE: * These values are given only for the benefit of calculating the end-of-life (EOL) RT_{NDT}.

NON-BELTLINE COMPONENT	MT'L TYPE OR WELD SEAM I.D.	HEAT/SLAB OR HEAT/LOT	HIGHEST REFERENCE TEMPERATURE RT_{NDT} (°F)
Shell Ring Connected to Vessel Flange	SA 533, GR.B, C1.1	All Heats	+19
Bottom Head Dome	SA 533, GR.B, C1.1	All Heats	+30
Bottom Head Torus	SA 533, GR.B, C1.1	All Heats	+30
LPCI Nozzles(1)	SA 508, C1.2	All Heats	-20
Top Head Torus	SA 533, GR.B, C1.1	All Heats	+19
Top Head Flange	SA 508, C1.2	All Heats	+10
Vessel Flange	SA 508, C1.2	All Heats	+10
Feedwater Nozzle	SA 508, C1.2	All Heats	-20
Weld Metal	All RPV Welds	All Heats	0
Closure Studs	SA 540, GR.B, 24	All Heats	Meet 45 ft-lbs & 25 mils lateral expansion at +10°F

(1) The design of the Hope Creek vessel results in these nozzles experiencing a predicted EOL fluence at 1/4T of the vessel thickness of 2.81×10^{17} n/cm^2. Therefore, these nozzles are predicted to have an EOL RT_{NDT} of +24°F.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 88 TO FACILITY OPERATING LICENSE NO. NPF-57

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By letter dated July 27, 1995, the Public Service Electric & Gas Company (the licensee) submitted a request for a change to the Hope Creek Generating Station (HCGS), Technical Specifications (TSs). The proposed Technical Specification (TS) revision would incorporate updated pressure vs. temperature operating limit curves contained in TS Figure 3.4.6.1-1. The revision would also changes TS Surveillance Requirement 4.4.6.1.3 based upon implementation of Regulatory Guide 1.99, Revision 2, in accordance with Generic Letter 88-11, for reactor vessel material surveillance specimens.

2.0 BACKGROUND

The new pressure and temperature (P/T) limits are a result of data obtained from the first set of specimen capsules removed during refueling outage 5. The new peak surface fluence estimate is about 14% greater than the previous estimate. Since this is the first set of specimen capsules removed, only the flux wire test results are factored into the beltline adjusted reference temperature (ART) calculations. The flux wire test results and the lead factor from the last fuel cycle were used to estimate the new 32 effective full power years (EFPY) fluence. The chemistry factor values described in Regulatory Guide (RG) 1.99, Revision 2, can be modified when two or more sets of credible surveillance capsule data, as defined in the RG, become available. The report supporting these changes was submitted to the NRC in Letter LR-N95071 dated June 9, 1995.

The staff evaluates the P/T limits based on the following NRC regulations and guidance: Appendix G to 10 CFR Part 50; Generic Letters 88-11 and 92-01; RG 1.99, Revision 2; and Standard Review Plan (SRP) Section 5.3.2. Appendix G to 10 CFR Part 50 requires that P/T limits for the reactor vessel must be at least as conservative as those obtained by Appendix G to Section III of the American Society of Mechanical Engineers (ASME) Code. GL 88-11 requests that licensees use the methods in RG 1.99, Revision 2, to predict the effect of neutron irradiation by calculating the ART of reactor vessel materials. The ART is defined as the sum of initial nil-ductility transition reference temperature (RT_{NDT}) of the material, the increase in RT_{NDT} caused by neutron irradiation, and a margin to account for uncertainties in the prediction method. The increase in RT_{NDT} is calculated from the product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the vessel material. GL 92-01 requests licensees to submit reactor vessel materials data, which the staff will use in the review of the P/T limits submittals.

SRP 5.3.2 provides guidance on calculation of the P/T limits using linear elastic fracture mechanics methodology specified in Appendix G to Section III of the ASME Code. The linear elastic fracture mechanics methodology postulates sharp surface defects that are normal to the direction of maximum stress and have a depth of one-fourth of the reactor vessel beltline thickness (1/4T) and a length of 1-1/2 the beltline thickness. The critical locations in the vessel for this methodology is the 1/4T and 3/4T locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

3.0 DISCUSSION

For the Hope Creek reactor vessel, the licensee determined that the intermediate plate (I.D. #3, Heat 5K3025), with 0.15% Cu, 0.71% Ni, and initial RT_{NDT} of 19°F is the limiting material for the 1/4T location. The licensee calculated an ART of 72.5°F.

The staff verified that copper and nickel contents and initial RT_{NDT} of the reactor vessel beltline materials agreed with those in the licensee's updated responses to GL 92-01 for Hope Creek. The staff used the material properties to perform independent calculations of the ART values for the beltline materials using RG 1.99, Revision 2. Based on the calculations, the staff verified that the licensee's calculated ART value for the limiting material is acceptable.

Substituting the ART of the limiting material into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for (1) hydrostatic or leak testing, (2) heatup by non-nuclear means, cooldown following a nuclear shutdown and low power physics tests, and (3) operations with a critical core other than low power physics tests satisfy the requirements in Paragraphs IV.A.2 and IV.A.3 of Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50, also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.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 RT_{NDT} of 10°F provided by the licensee, the staff has determined that the proposed P/T limits have satisfied the requirement for the closure flange region during normal operation, hydrostatic pressure test and leak test.

4.0 EVALUATION

The staff has performed an independent analysis to verify the licensee's proposed P/T limits. The staff concludes that the proposed P/T limits for (1) hydrostatic or leak testing, (2) heatup by non-nuclear means, cooldown following a nuclear shutdown and low power physics tests, and (3) operations with a critical core other than low power physics tests are valid for 32 EFPY, because: 1) the limits conform to the requirements of Appendix G of 10 CFR Part 50 and GL 88-11, and 2) the material properties and chemistry used in calculating the P/T limits are consistent with or conservative compared to data submitted under GL 92-01; hence, the proposed P/T limits may be incorporated in the Hope Creek Generating Station TS 3/4.4.6.1. The licensee has also proposed deletion of the specific material surveillance methodology of TS 4.4.6.1.3a and b. Deletion of TS surveillance requirement 4.4.6.1.3a and b is acceptable because the licensee will utilize the calculation methodology in RG 1.99, Revision 2 to calculate embrittlement. The deletion does not affect the surveillance program. Hence, the licensee's surveillance program must still comply with 10 CFR Part 50, Appendix H. In addition, the proposed changes in the Bases section of the TS are consistent with the P/T limits changes and; therefore, are acceptable.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (60 FR 47624). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: A. Lee

Date: November 28, 1995

8.0 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. Code of Federal Regulations, Title 10, Part 50, Appendix G, Fracture Toughness Requirements

4. Generic Letter 88-11, NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations, July 12, 1988

5. ASME Boiler and Pressure Vessel Code, Section III, Appendix G for Nuclear Power Plant Components, Division 1, "Protection Against Nonductile Failure"

6. July 27, 1995, Letter from J. J. Hagan to USNRC Document Control Desk, Request for License Amendment Hope Creek Generating Station

7. R. G. Carey, Hope Creek 1 Generating Station RPV Surveillance Materials Testing and Fracture Toughness Analysis, GE-NE-523-A164-1294, General Electric Company, April 1995