December 16, 1991

Docket No. 50-354

Mr. Steven E. Miltenberger
Vice President and Chief Nuclear
Officer
Public Service Electric & Gas Company
Post Office Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. Miltenberger:

SUBJECT: GENERIC LETTER 91-01 LICENSE AMENDMENT, HOPE CREEK GENERATING STATION (TAC NO. M81927)

The Commission has issued the enclosed Amendment No. 46 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 October 10, 1991.

This amendment removes TS Table 4.4.6.1.3-1, "Reactor Vessel Material Surveillance Program - Withdrawal Schedule," from the TS. References to the table in TS 3/4.4.6 and the associated Bases are also being deleted.

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

Sincerely,

/S/
Stephen Dembek, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 46 to License No. NPF-57
2. Safety Evaluation

cc w/enclosures:
See next page

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

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Mr. Steven E. Miltenberger
Public Service Electric & Gas Co.

Hope Creek Generating Station

cc:

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 46
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 October 10, 1991 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. 46 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PSE&G shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Charles L. Miller

Charles L. Miller, 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: December 16, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 46

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. Overleaf pages provided to maintain document completeness.*

Remove	Insert
xi	xi
xii	xii*
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 4-24	3/4 4-24
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*
B 3/4 4-8	B 3/4 4-8

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curves C and C' 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 Figure 3.4.6.1-1 based on the greater of the following criteria:

a. The actual shift in reference temperature for plate material from heat 5K3238-1 and weld metal 510-01205 as determined by Charpy impact test, or

b. The predicted shift in reference temperatures for plate material from heat 5K3025-1 as determined by Regulatory Guide 1.99, "Radiation Damage to Reactor Vessel Materials."

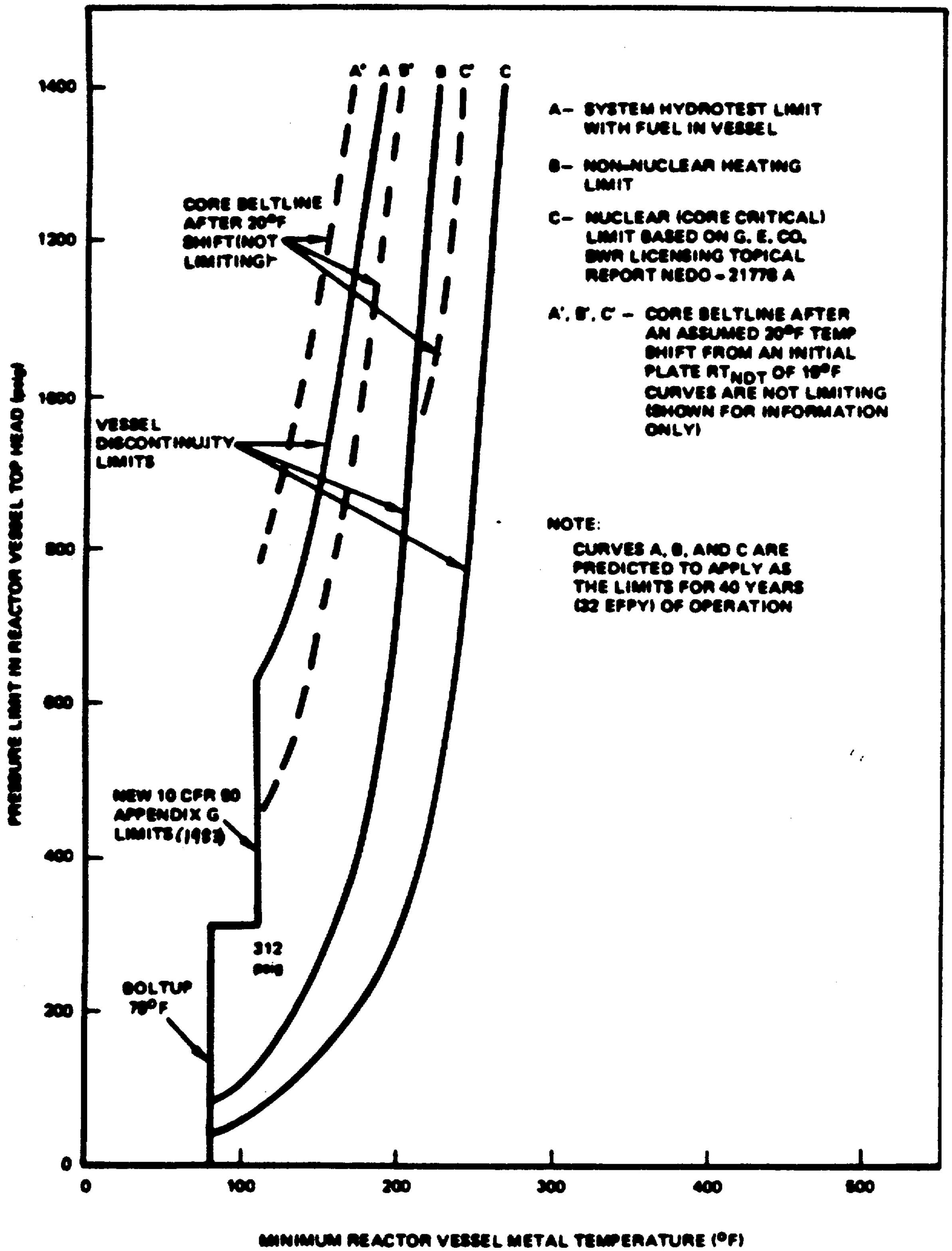
4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 70°F:

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

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

   2. $\leq$ 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.



MINIMUM REACTOR PRESSURE VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE

Figure 3.4.6.1-1

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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 (4.9) 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.

The operating limit curves of Figure 3.4.6.1-1 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 FSAR 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 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, phosphorus 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, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The pressure/ temperature limit curve, Figure 3.4.6.1-1, curves A', B' and C', 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 Figure 3.4.6.1-1 shall be adjusted, as required, on the basis of the flux wire data and recommendations of Regulatory Guide 1.99, Revision 1.

PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown in Figures 3.4.6.1-1, curves C, and C', and A and A', for reactor criticality and for inservice leak and hydrostatic testing 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(%)	P(%)	HIGHEST RT_{NDT}(°F)	PREDICTED $\Delta\ RT_{NDT}$(°F)	UNIRRADIATED UPPER SHELF (FT-LBS)	MAX. EOL RT_{NDT}(°F)
Plate	SA-533 GR B CL.1	5K3025-1	.15	.012	+19	20	76	+39
Weld	Long. seams for shells 4&5 and girth weld between 4&5	D55040/ 1125-02000	.08	.010	-30	17	135	-13

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, Cl.1	All Heats	+19
Bottom Head Dome	SA 533, GR.B, Cl.1	All Heats	+30
Bottom Head Torus	SA 533, GR.B, Cl.1	All Heats	+30
LPCI Nozzles	SA 508, Cl.2	All Heats	-20
Top Head Torus	SA 533, GR.B, Cl.1	All Heats	+19
Top Head Flange	SA 508, Cl.2	All Heats	+10
Vessel Flange	SA 508, Cl.2	All Heats	+10
Feedwater Nozzle	SA 508, Cl.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

The design of the Hope Creek vessel results in these nozzles experiencing a predicted EOL fluence at 1/4T of the vessel thickness of 1.6 x 10^{17} n/cm^2. Therefore, these nozzles are predicted to have an EOL RT_{NDT} of -6°F.

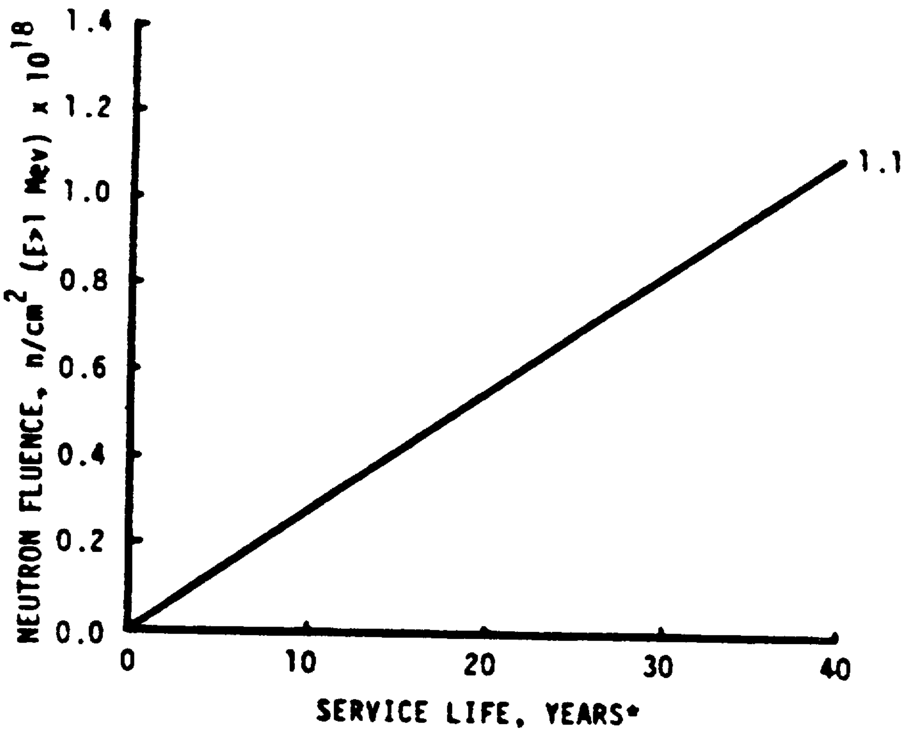

FIGURE B 3/4 4.6-1 FAST NEUTRON FLUENCE (E>1 Mev) AT 1/4 T AS A FUNCTION OF SERVICE LIFE*

Bases Figure B 3/4.4.6-1

* At 90% of RATED THERMAL POWER and 90% availability

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 46 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 October 10, 1991, the Public Service Electric & Gas Company and Atlantic City Electric Company (the licensees) submitted a request for changes to the Hope Creek Generating Station, Technical Specifications (TS). The proposed change removes TS Table 4.4.6.1.3-1 providing the schedule for reactor vessel material specimen withdrawal. Guidance on the proposed TS change was provided by Generic Letter 91-01, of January 4, 1991, to all holders of operating licenses or construction permits for nuclear power reactors.

2.0 EVALUATION

Technical Specification 3/4.4.6, "Pressure/Temperature Limits," contains a limiting condition for operation for the Reactor Coolant System (RCS) that limits the rate of change in temperature and pressure to values consistent with the fracture toughness requirements of the American Society of Mechanical Engineers' (ASME) Code and Appendix G to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50). Changes in the values of these limits are necessary because the fracture toughness properties of ferritic materials in the reactor vessel change as a function of the reactor operating time (neutron fluence).

For this reason, the TS include a surveillance requirement, TS 4.4.6.1.3, to require the removal and examination of the irradiated specimens of reactor vessel material. The licensees examine the specimens to determine the changes in material properties in accordance with the requirements of Appendix H to 10 CFR Part 50. Table 4.4.6.1.3-1 identifies the material specimens and specifies the schedule for removal of each specimen.

The removal of the schedule for withdrawing material specimens from the TS will eliminate the necessity of a license amendment to make changes to this schedule. However, Section II.B.3 of Appendix H to 10 CFR Part 50 requires the submittal of a proposed withdrawal schedule for material specimens to the U.S. Nuclear Regulatory Commission (NRC) and approval by the NRC before implementation. Hence, adequate regulatory controls exist to control changes to this schedule.

The generic letter requires that a licensee proposing to delete the withdrawal schedule from the TSs commit to maintaining the NRC-approved version of the schedule in the Updated Final Safety Analysis Report (UFSAR). Based on review of the Hope Creek UFSAR, the licensees determined and the staff verified that all of the information contained in TS Table 4.4.6.1.3-1 is already included in the UFSAR. The withdrawal schedule is included on Page 5.3-11 and Page 5A-10 while the lead factors and vessel locations are contained on Page 5A-9. In addition, the licensees will include any subsequent NRC-approved revisions to this schedule in an update of the UFSAR. The inclusion of the withdrawal schedule in the UFSAR provides a source for this information that is readily available as a reference for NRC inspectors and other staff use. Finally, the surveillance requirements for removing material specimens and the bases section for this specification remain unchanged except for the removal of the reference to Table 4.4.6.1.3-1.

The licensees have proposed a change to TS 3/4.4.6 that is consistent with the guidance provided in Generic Letter 91-01 for removal of Table 4.4.6.1.3-1 from the TS. The NRC has reviewed this matter and finds that the proposed changes to the TS for the Hope Creek Generating Station are acceptable.

Additonally, the multiplier for the values of neutron fluence on the ordinate of Bases Figure B 3/4.4.6-1 is incorrect. Since the proposed change corrects an editorial error in the existing TS Bases, the change is purely administrative in nature and is acceptable.

In addition to the changes requested by the licensees, the staff, with the concurrence of the licensees, made the following editorial changes to the TS:

The words "This page intentionally left blank" were added to page 3/4 4-24. This page had previously contained Table 4.4.6.1.3-1.

The word "(Deleted)" was substituted for the table description on index page xi after the reference to Table 4.4.6.1.3-1.

The two above changes were administrative in nature and did not increase the scope of the original amendment request and did not affect the staff's original no significant hazards determination.

3.0 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.

4.0 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. The NRC staff has determined that the 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 the amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 57702). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). The staff made this determination because the removal of the schedule for removing material specimens from the TS does not alter the necessity for formal NRC approval of changes to the schedule as established by Section II.B.3 of Appendix H to 10 CFR Part 50. 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.

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

Principal Contributors: T. Dunning, OTSB/DOEA
S. Dembek, PDI-2/DRPE

Date: December 16, 1991