March 6, 1997

Mr. Michael B. Roche
Vice President and Director
GPU Nuclear Corporation
Oyster Creek Nuclear Generating Station
P.O. Box 388
Forked River, NJ 08731

SUBJECT: ISSUANCE OF AMENDMENT TO UPDATE THE PRESSURE-TEMPERATURE LIMITS CURVES FOR THE OYSTER CREEK NUCLEAR GENERATING STATION REACTOR PRESSURE VESSEL (TAC NO. M96405)

Dear Mr. Roche:

The Commission has issued the enclosed Amendment No. 188to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated August 23, 1996 (TSCR 245), as supplemented January 8, 1997.

The amendment updates the pressure-temperature limits up to 22, 27, and 32 effective full power years (EFPY).

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

(Original Signed By)

Ronald B. Eaton, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-219

Enclosures: 1. Amendment No.188 to DPR-16
2. Safety Evaluation

cc w/encls: See next page

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

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Vice President and Director
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cc w/encls: See next page

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

GPU NUCLEAR CORPORATION

AND

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 188
License No. DPR-16

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

   A. The application for amendment by GPU Nuclear Corporation, et al., (the licensee) dated August 23, 1996, as supplemented January 8, 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-16 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 188, are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Patrick D. Milano, Acting Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 6, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 188

FACILITY OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace the following pages of the Appendix A, Technical Specifications, with the attached pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
3.3-1	3.3-1
3.3-5	3.3-5
3.3-8a	3.3-8a
3.3-9	--
3.3-9a	3.3-9a
3.3-9b	3.3-9b
3.3-9c	3.3-9c
4.3-1	4.3-1
4.3-2	4.3-2

3.3 REACTOR COOLANT

Applicability: Applies to the operating status of the reactor coolant system.

Objective: To assure the structure integrity of the reactor coolant system.

Specification: A. Pressure Temperature Relationships

(i) Reactor Vessel Pressure Tests - the minimum reactor vessel temperature at a given pressure shall be in excess of that indicated by the curve A in Figures 3.3.1, 3.3.2 and 3.3.3 for reactor operations to 22, 27 and 32 effective full power years, respectively. The maximum temperature for Reactor Vessel Pressure Testing is 250°F.

(ii) Heatup and Cooldown Operations: Reactor noncritical -- the minimum reactor vessel temperature for heatup and cooldown operations at a given pressure when the reactor is not critical shall be in excess of that indicated by the curve B in Figures 3.3.1, 3.3.2 and 3.3.3 for reactor operations up to 22, 27 and 32 effective full power years, respectively.

(iii) Power operations -- the minimum reactor vessel temperature for power operations at a given pressure shall be in excess of that indicated by the curve C in Figures 3.3.1, 3.3.2 and 3.3.3 for reactor operations up to 22, 27 and 32 effective full power years respectively.

Note: Curves A, B and C in Figures 3.3.1, 3.3.2 and 3.3.3 apply when the closure head is on the reactor vessel and studs are fully tensioned.

(iv) Appropriate new pressure temperature limits must be generated when the reactor system has reached thirty two (32) effective full power years of reactor operation.

B. Reactor Vessel Closure Head Boltdown: The reactor vessel closure head studs may be elongated .020" (1/3 design preload) with no restrictions on reactor vessel temperature as long as the reactor vessel is at atmospheric pressure. Full tensioning of the studs is not permitted unless the temperature of the reactor vessel flange and closure head flange is in excess of 85°F.

C. Thermal Transients

1. The average rate of reactor coolant temperature change during normal heatup and cooldown shall not exceed 100°F in any one hour period.

2. The pump in an idle recirculation loop shall not be started unless the temperature of the coolant within the idle recirculation loop is within 50°F of the reactor coolant temperature.

Transformation temperature. The minimum temperature for pressurization at any time in life has to account for the toughness properties in the most limiting regions of the reactor vessel, as well as the effects of fast neutron embrittlement.

Curves A, B and C on Figures 3.3.1, 3.3.2 and 3.3.3 are derived from an evaluation of the fracture toughness properties performed on the specimens contained in Reactor Vessel Materials Surveillance Program Capsule No. 2 (Reference 14). The results of dosimeter wire analyses (Reference 14) indicated that the neutron fluence (E>1.0 MeV) at the end of 32 effective full power years of operation is 2.36 x 10^{18} n/cm^2 at the 1/4T (T=vessel wall thickness) location. This value was used in the calculation of the adjusted reference nil-ductility temperature which, in turn, was used to generate the pressure-temperature curves A, B and C on Figures 3.3.1, 3.3.2 and 3.3.3 (Reference 15). The 250°F maximum pressure test temperature provides ample margin against violation of the minimum required temperature. Secondary containment is not jeopardized by a steam leak during pressure testing, and the Standby Gas Treatment system is adequate to prevent unfiltered release to the stack.

Stud tensioning is considered significant from the standpoint of brittle fracture only when the preload exceed approximately 1/3 of the final design value. No vessel or closure stud minimum temperature requirements are considered necessary for preload values below 1/3 of the design preload with the vessel depressurized since preloads below 1/3 of the design preload result in vessel closure and average bolt stresses which are less than 20% of the yield strengths of the vessel and bolting materials. Extensive service experience with these materials has confirmed that the probability of brittle fracture is extremely remote at these low stress levels, irrespective of the metal temperature.

The reactor vessel head flange and the vessel flange in combination with the double "O" ring type seal are designed to provide a leak tight seal when bolted together. When the vessel head is placed on the reactor vessel, only that portion of the head flange near the inside of the vessel rests on the vessel flange. As the head bolts are replaced and tensioned, the vessel head is flexed slightly to bring together the entire contact surface adjacent to the "O" rings of the head and vessel flange. The original Code requirement was that boltup be done at qualification temperatures (T3OL) plus 60°F. Current Code requirements state (Ref. 16) that for application of full bolt preload and reactor pressure up to 20% of hydrostatic test pressure, the RPV metal temperature must be at RT_{NDT} or greater. The boltup temperature of 85°F was derived by determining the highest value of (T3OL + 60) and the highest value of RT_{NDT}, and by choosing the more conservative value of the two. Calculated values of (T3OL + 60) and RT_{NDT} of the RPV metal temperature were 85°F and 36°F, respectively (Ref. 15). Therefore, selecting the boltup temperature to be 85°F provides 49°F margin over the current Code requirement based on RT_{NDT}.

Detailed stress analyses(4) were made on the reactor vessel for both steady state and transient conditions with respect to material fatigue. The results of these analyses are presented and compared to allowable stress limits in Reference (4). The specific conditions analyzed included 120 cycles of normal startup and shutdown with a heating and cooling rate of 100°F per hour applied continuously over a temperature range of 100°F to 546°F and for 10 cycles of emergency cooldown at a rate of 300°F per hour applied over the same range. Thermal stresses from this analysis combined with the primary load

References:

(1) FDSAR, Volume I, Section IV-2
(2) Letter to NRC dated May 19, 1979, "Transient of May 2, 1979"
(3) General Electric Co. Letter G-EN-9-55, "Revised Natural Circulation Flow Calculation", dated May 29, 1979
(4) Licensing Application Amendment 16, Design Requirements Section
(5) (Deleted)
(6) FDSAR, Volume I, Section IV-2.3.3 and Volume II, Appendix H
(7) FDSAR, Volume I, Table IV-2-1
(8) Licensing Application Amendment 34, Question 14
(9) Licensing Application Amendment 28, Item III-B-2
(10) Licensing Application Amendment 32, Question 15
(11) (Deleted)
(12) (Deleted)
(13) Licensing Application Amendment 16, Page 1
(14) GPUN TDR 725 Rev. 3: Testing and Evaluation of Irradiated Reactor Vessel Materials Surveillance Program Specimens
(15) GENE-B13-01769 (GE Nuclear Energy): Pressure-Temperature Curves Per Regulatory Guide 1.99, Revision 2 for Oyster Creek Nuclear Generating Station.
(16) Paragraph G-2222(C), Appendix G, Section XI, ASME Boiler and Pressure Vessel Code, 1989 Edition with 1989 Addenda, "Fracture Toughness Criteria for Protection Against Failure."

FIGURE 3.3.1

OYSTER CREEK P-T CURVES VALID TO 22 EFPY

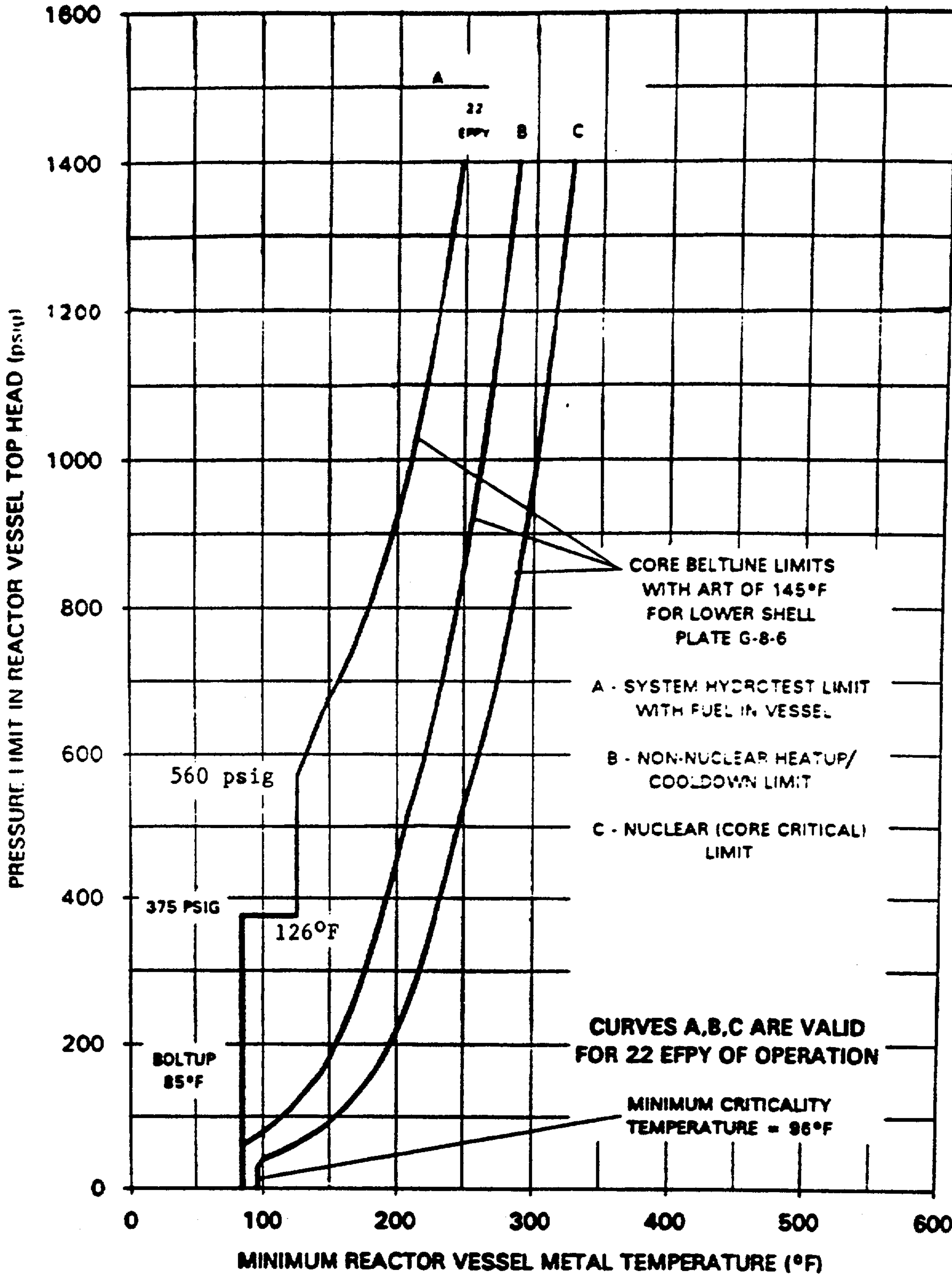


FIGURE 3.3.2

OYSTER CREEK P-T CURVES VALID TO 27 EFPY

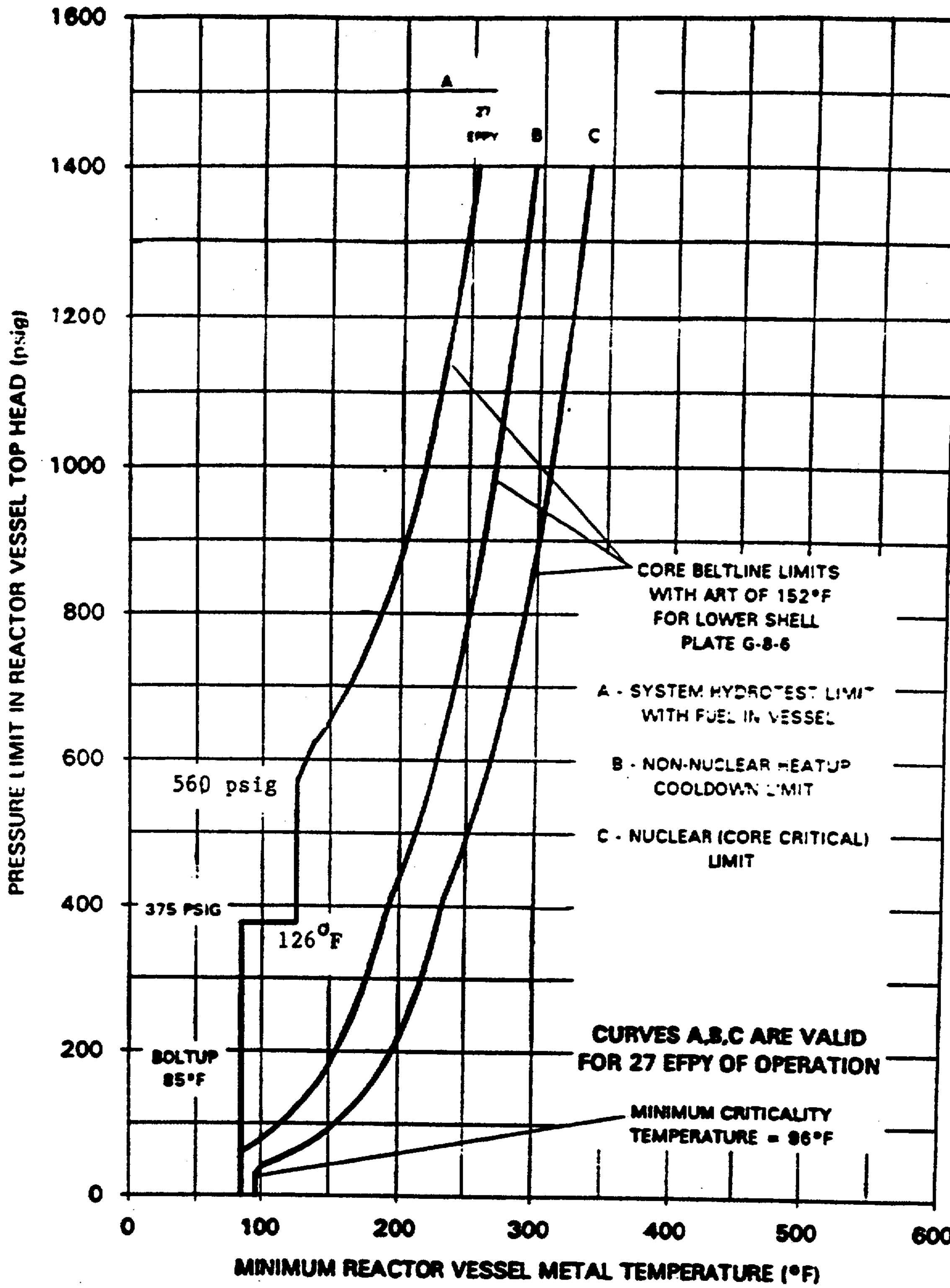
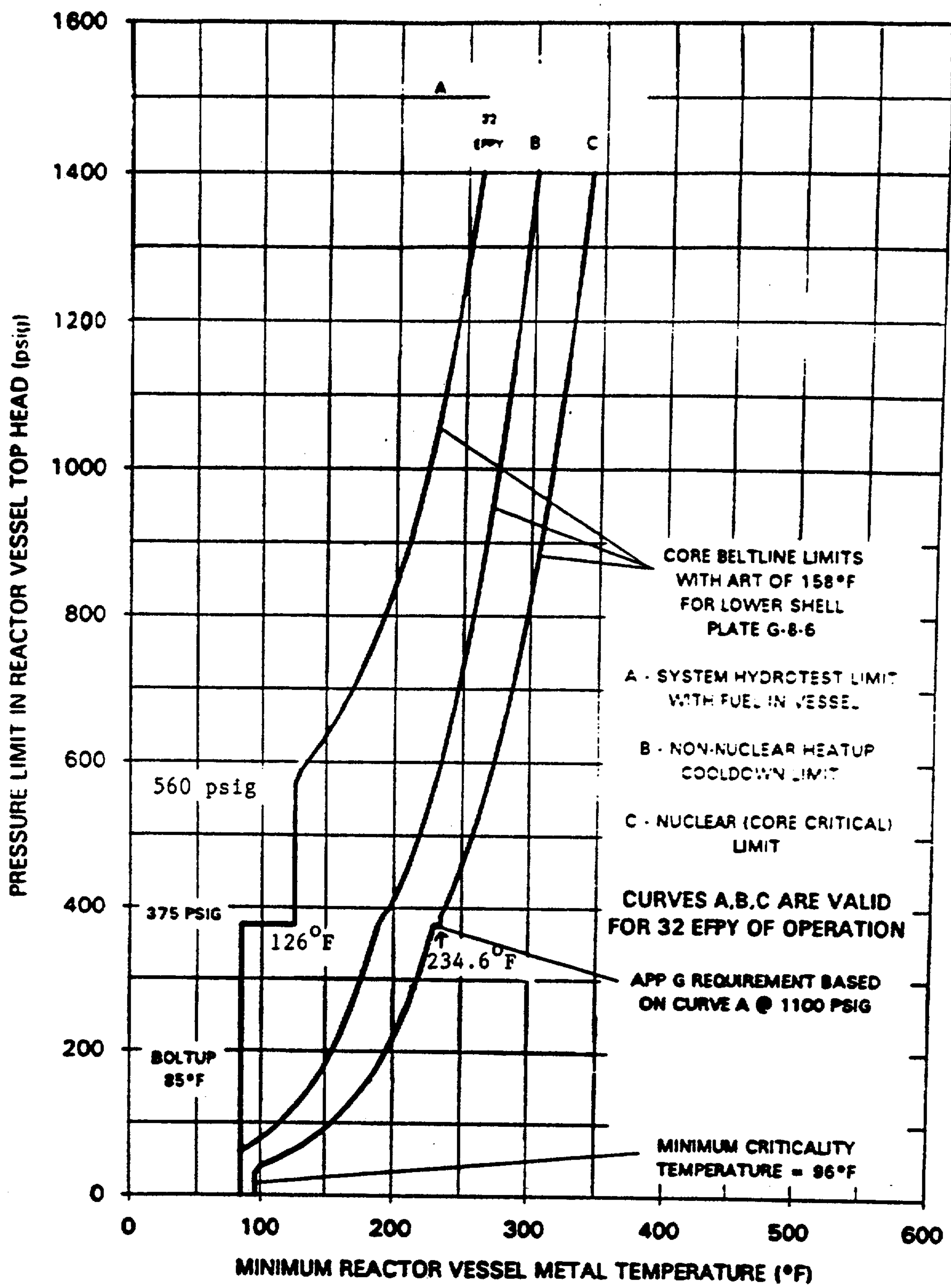


FIGURE 3.3.3

OYSTER CREEK P-T CURVES VALID TO 32 EFPY



4.3 REACTOR COOLAN'

Applicability: Applies to the surveillance requirements for the reactor coolant system.

Objective: To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Materials surveillance specimens and neutron flux monitors shall be installed in the reactor vessel adjacent to the wall at the midplane of the active core. Specimens and monitors shall be periodically removed, tested, and evaluated to determine the effects of neutron fluence on the fracture toughness of the vessel shell materials. The results of these evaluations shall be used to assess the adequacy of the P-T curves A, B and C in Figures 3.3.1, 3.3.2 and 3.3.3. New curves shall be generated as required.

B. Inservice inspection of ASME Code Class 1, Class 2 and Class 3 systems and components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR, Section 50.55a(g)(6)(i).

C. Inservice testing of ASME Code Class 1, Class 2 and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR, Section 50.55a(g)(6)(i).

D. A visual examination for leaks shall be made with the reactor coolant system at pressure during each scheduled refueling outage or after major repairs have been made to the reactor coolant system in accordance with Article 5000, Section XI. The requirements of specification 3.3.A shall be met during the test.

E. Each replacement safety valve or valve that has been repaired shall be tested in accordance with subsection IWV-3510 of Section XI of the ASME Boiler and Pressure Vessel Code. Setpoints shall be as follows:

Number of Valves	Set Points (psig)
4	1212 ± 12
5	1221 ± 12

F. A sample of reactor coolant shall be analyzed at least every 72 hours for the purpose of determining the content of chloride ion and to check the conductivity.

* G. :mary Coolant System Pressure Isolatic 'alves Specification:

1. Periodic leakage testing [a] on each valve listed in Table 4.3.1 shall be accomplished prior to exceeding 600 psig reactor pressure every time the plant is placed in the cold shutdown condition for refueling, each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, whenever the valve is moved whether by manual actuation or due to flow conditions, and after returning the valve to service after maintenance, repair or replacement work is performed.

H. Reactor Coolant System Leakage

1. Unidentified leakage rate shall be calculated at least once every 4 hours.

2. Total leakage rate (identified and unidentified) shall be calculated at least once every 8 hours.

3. A channel calibration of the primary containment sump flow integrator and the primary containment equipment drain tank flow integrator shall be conducted at least once per 18 months.

I. An inservice inspection program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the NRC staff positions on schedule, methods, personnel, and sample expansion included in the generic letter or in accordance with alternate measures approved by the NRC staff.

Bases:

Data is available relating neutron fluence (E>1.0MeV) and the change in the Reference Nil-Ductility Transition Temperature (RT_{NDT}). The pressure-temperature (P-T) operating curves A, B, and C in Figures 3.3.1, 3.3.2, and 3.3.3 were developed based on the results of testing and evaluation of specimens removed from the vessel after 8.38 EFPY of operation. Similar testing and analysis will be performed throughout vessel life to monitor the effects of neutron irradiation on the reactor vessel shell materials.

The inspection program will reveal problem areas should they occur, before a leak develops. In addition, extensive visual inspection for leaks will be made on critical systems. Oyster Creek was designed and constructed prior to

[a] To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

* NRC Order dated April 20, 1981.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 188

TO FACILITY OPERATING LICENSE NO. DPR-16

GPU NUCLEAR CORPORATION AND

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated August 23, 1996, General Public Utilities (GPU) Nuclear Corporation (the licensee) submitted a request for changes to the Oyster Creek Nuclear Generating Station (OCNGS) Technical Specifications (TS). GPU proposed to revise TS 3.3.A (i), (ii), (iii), and (iv); the pressure-temperature (P-T) limit curves in Figure 3.3.1; TS 4.3.A; and the Bases and references for the aforementioned sections. GPU supplemented this submittal by letter dated January 8, 1997, in response to a request for additional information (RAI) from the NRC staff of November 27, 1996. The January 8, 1997, letter provided clarifying information within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination.

The staff has reviewed the information provided by the licensee and has determined that the licensee used methodologies consistent with the requirements of Title 10 of the Code of Federal Regulations Part 50 (10 CFR Part 50), Appendix G, Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Code, and Standard Review Plan (SRP) Section 5.3.2, "Pressure-Temperature Limits."

The requirements and guidelines that have been established or endorsed by the NRC concerning pressure versus temperature limitations on the reactor vessel environment provide adequate margins of safety against brittle fracture of the reactor vessel during all modes of operation. Appendix G of 10 CFR Part 50 identifies three specific vessel operating conditions to consider when developing P-T limit curves: vessel hydrostatic or leak-rate testing; normal operation with the core not critical, and normal operation with the core critical. The licensee's submittal addressed each of these operating conditions and proposed a P-T limit curve for each.

All regions of the reactor vessel must be evaluated to ensure vessel integrity during each of the operating conditions noted above. As addressed in 10 CFR Part 50, Appendix G, two regions of the reactor vessel are specifically identified relating to the establishment of these limitations: the closure flange region and the vessel beltline region. In addition, Appendix G to Section XI of the ASME Code recommends methodologies for the analysis of geometrically complex vessel regions, namely the vessel lower head and the feedwater nozzle region. The analysis submitted by the licensee considered each of these regions. The licensee proposed that the P-T limit curve for each operational condition then be constructed as a composite of the most limiting values for all pressures of interest. This resulted in the P-T limit curves for each operational condition being controlled at low pressures by consideration of the reactor vessel flange region and at high pressures by consideration of the reactor vessel beltline. The reactor vessel bottom head was the controlling region for vessel hydrostatic/leak rate testing at intermediate pressures. The feedwater nozzle region controlled both of the normal operation curves in the intermediate pressure regime.

The regulatory requirements concerning the development of P-T Limits are contained in Appendix G to 10 CFR Part 50 and provide the general basis for these limits. The requirements of 10 CFR Part 50, Appendix G, specifically state that the P-T limits for a vessel must be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of Appendix G to Section XI of the ASME Code. The staff has also established guidance for the development of P-T limits in SRP Section 5.3.2. Staff guidance on the calculation of material embrittlement as a result of neutron radiation has been published in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations."

2.0 EVALUATION

The staff first reviewed the nil-ductility reference temperatures (RT_{ndt}) proposed by the licensee for the reactor vessel non-beltline regions and determined that they were acceptable. Due to low neutron fluence levels these materials do not experience any irradiation-induced shift in RT_{ndt} through 32 effective full power years (EFPYs). The staff then confirmed that the values of the adjusted RT_{ndt} (a combination of the initial RT_{ndt}, the irradiation-induced shift in RT_{ndt} and a margin term) for the limiting beltline material (Plate G-8-6) were 145, 152, and 158 °F as calculated by the licensee at 22, 27, and 32 EFPYs, respectively. The shift in RT_{ndt} for the beltline materials was calculated by the staff in accordance with Regulatory Guide 1.99, Revision 2.

With this information, the staff was able to verify the licensee's P-T limit curves using the methodology of SRP Section 5.3.2 and Appendix G to Section XI of the ASME Code. Initially the staff calculated the P-T limit curves for the limiting beltline material at 22, 27, and 32 EFPYs for vessel hydrostatic/leak rate testing and for the limiting normal operation transient, a 100 °F/hr cooldown. The staff then reviewed the licensee's calculations for the non-beltline lower head and feedwater nozzle regions from the RAI response and verified those results. Finally, the staff constructed the composite P-T curves for each operational condition from the information noted above and the minimum temperature constraints based on the limiting RT_{ndt} of the closure flange region as required by 10 CFR Part 50, Appendix G.

The staff determined that the P-T limits proposed for vessel hydrostatic/leak rate testing by the licensee were acceptable to 22, 27, and 32 EFPYs. The staff also confirmed that the P-T limits curves for 22, 27, and 32 EFPYs submitted by the licensee for normal operation under core critical and non-critical conditions were consistent with those generated using the methodologies of SRP Section 5.3.2 and Appendix G of the ASME Code. In addition, normal boiling water reactor operations are constrained to saturated water conditions, and these conditions are very conservative with respect to the normal operation limits as calculated by the acceptable methodologies listed above.

The staff has reviewed the information provided by the licensee in the documents listed above. The staff has determined that the licensee used methodologies consistent with the 10 CFR Part 50, Appendix G, Appendix G to Section XI of the ASME Code, Regulatory Guide 1.99, Revision 2, and SRP Section 5.3.2. Therefore, these changes to the Oyster Creek Nuclear Generating Station TS are acceptable.

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 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 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 (61 FR 47977). 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.

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 Contributor: M. Mitchell

Date: March 6, 1997