Docket
File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
April 11, 1991

Docket No. 50-219

Mr. John J. Barton, Director
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Barton:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 79374)

The Commission has issued the enclosed Amendment No. 151 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated January 11, 1991, as supplemented March 12, 1991.

The amendment revises Technical Specification (TS) 3.3 of the Oyster Creek Nuclear Generating Station TS. Specifically, the amendment revises TS Section 3.3.A, pressure/temperature limits of the reactor coolant system for operation up to 17 effective full power years. It also revises TS Section 3.3.B to provide a new reactor vessel temperature limit for full tensioning of the reactor vessel closure head studs.

As a separate issue, GPU Nuclear Corporation (GPUN) could not provide unirradiated Charpy upper shelf energy values for all three reactor vessel beltline welds-86054B/4B5F, 86054B/4D4F, and 1248/4M2F. Therefore, we request that GPUN provide the unirradiated upper shelf energy of the welds within 6 months of the issuance of the amendment.

We request that you respond within 30 days of receipt of this letter indicating your intent to comply with the staff's requirements.

This requirement affects fewer than 10 respondents and, therefore, is not subject to Office of Management and Budget review under P.L. 511.

A copy of the related Safety Evaluation is enclosed. Also enclosed is the Notice of Issuance which has been forwarded to the Office of the Federal Register for publication.

Sincerely,

/s/

Alexander W. Dromerick, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 151to DPR-16
2. Safety Evaluation

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See next page

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

GPU NUCLEAR CORPORATION

AND

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 151
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 January 11, 1991, as supplemented March 12, 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;

   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 Provisional 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. 151, 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

John F. Stolz

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 11, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 151

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

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

Remove	Insert
Page 3.3-1	Page 3.3-1
3.3-5	3.3-5
3.3-6	3.3-6*
3.3-7	3.3-7*
3.3-8	3.3-8*
3.3-8a	3.3-8a
3.3-9	3.3-9
4.3-1	4.3-1
4.3-2	4.3-2

*No change - Overflow page

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 Figure 3.3.1. 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 Figure 3.3.1.

(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 Figure 3.3.1.

Note: Curves (a),(b) and (c) in Figure 3.3.1 apply when the closure head is on the reactor vessel and studs are fully tensioned.

(iv) Appropriate new pressure temperature limits must be approved as part of this Technical Specification when the reactor system has reached seventeen (17) 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.

Transformatic :emperature. The minimum temp ture 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 Figure 3.3.1 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 17 effective full power years of operation is 1.25×10^{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 Figure 3.3.1 (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 stresses fall within ASME Code Section III allowable stress intensities. Although the Oyster Creek Unit 1 reactor vessel was built in accordance with Section I of the ASME Code, the design criteria included in the reactor vessel specifications were in essential agreement with the criteria subsequently incorporated into Section III of the Code.(6)

The expected number of normal heatup and cooldown cycles to which the vessel will be subjected is 80(7). Although no heatup or cooldown rates of 300°F per hour are expected over the life the vessel and the vessel design did not consider such events(6), stress analyses have been made which showed the allowable number of such events is 22,000 on the basis of ASME Section III alternating stress limits.

During reactor operation, the temperature of the coolant in an idle recirculation loop is expected to remain at reactor coolant temperature unless it is valved out of service. Requiring the coolant temperature in an idle loop to be within 50°F of the reactor coolant temperature before the sump is started assures that the change in coolant temperature at the reactor vessel nozzles and bottom head region are within the conditions analyzed for the reactor vessel as discussed above.

Allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to makeup coolant system leakage in the event of loss of offsite AC power. The normally expected background leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work(8) utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in the 3.3-D on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm as specified in 3.3-D, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage of the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the plant should be shut down to allow further investigation and corrective action.

The drywell floor drain sump and equipment drain tank provide the primary means of leak detection(9,10). Identified leakage is that from valves and pumps in the reactor system and from the reactor vessel head flange gasket. Leakage through the seals of this equipment is piped to the drywell equipment drain tank. Leakage from other sources is classified as unidentified leakage and is collected in the drywell floor drain sump. Leakage which does not flash in a vapor will drain in the sump. The vapor will be condensed in the drywell ventilation system and routed to the sump.

Condensate cannot leave the sump or the drywell equipment drain tank unless the respective pumps are running. The sump and the drain tank are provided with two pumps each. Alarms are provided for the sump that will actuate on a predetermined pumpout rate(10) and will be set to actuate at a leakage that is less than the unidentified leakage limit of 5 gpm.

Additional qualitative information(10) is available to the operator via the monitored drywell atmospheric condition. However, this information is not quantitative since fluctuation in atmospheric conditions are normally expected, and quantitative measurements are not possible. The temperature of the closed cooling water which serves as coolant for the drywell ventilation system is monitored and also provides information which can be related to reactor coolant system leakage(9). Additional protection is provided by the drywell high pressure scram which would be expected to be reached within 30 minutes of a steam leak of about 12 gpm(10).

During a loss of offsite AC power, the control rod drive hydraulic pumps, which are powered by the diesels, each can supply 110 gpm water makeup to the reactor vessel. A 25 gpm limit for total leakage, identified and unidentified, was established to be less than the 110 gpm makeup of a single rod drive hydraulic pump to avoid the use of the emergency core cooling system in the event of a loss of normal AC power.

Materials in the primary system are primarily 304 stainless steel and zircaloy fuel cladding. The reactor water chemistry limits are placed upon conductivity and chloride concentration since conductivity is measured continuously and gives an indication of abnormal conditions or the presence of unusual materials in the coolant, while chloride limits are specified to prevent stress corrosion cracking of stainless steel.

Chlorides are known to (1) promote intergranular stress corrosion cracking of sensitized steels, (2) induce transgranular cracking of non-sensitized stainless steels, (3) promote pitting and (4) promote crevice attack in most RCS materials (BWR Water Chemistry Guidelines, EPRI, April 1, 1984). The higher the concentration, the faster the attack. Therefore, the level of chloride in the reactor water should be kept as low as is practically achievable. The limits are therefore set to be consistent with Regulatory Guide 1.56 (Rev. 1).

In the case of BWR's where no additives are used in the primary coolant, and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. When the conductivity is within its proper normal range, pH, chloride, and other impurities affecting conductivity and water quality must also be within their normal ranges. Significant changes in conductivity provide the operator with a warning mechanism so that he can investigate and remedy the conditions causing the change.

Measurements of pH, chloride, and other chemical parameters are made to determine the cause of the unusual conductivity and instigate proper corrective action. These can be done before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Several techniques are available to correct off-standard reactor water quality conditions including removal of impurities from reactor water by the cleanup system, reducing input of impurities causing off-standard conditions by reducing power and reducing the reactor coolant temperature to less than 212°F. The major benefit of reducing the reactor coolant temperature to less than 212°F is to reduce the temperature dependent corrosion rates and thereby provide time for the cleanup system to re-establish proper water quality.

Specifications 3.3.F.1 and 3.3.F.2 require a minimum of four OPERATING recirculation loops during reactor POWER OPERATION. Core parameters have not been established for POWER OPERATION with less than four OPERATING loops. Therefore, Specification 3.3.F.3 requires reactor POWER OPERATION to be terminated and the reactor placed in the REFUEL MODE or SHUTDOWN CONDITION within 12 hours.

During four loop POWER OPERATION the idle loop, when it is not isolated, is required to have its discharge valve closed and its discharge bypass and suction valves open. This provides and limits reactor coolant backflow through an idle loop and thus minimizes the occurrence of a severe cold water addition transient during startup of an idle loop. In addition, with the discharge bypass and suction valves in an idle loop open the coolant inventory in the loop is available during LOCA blowdown.

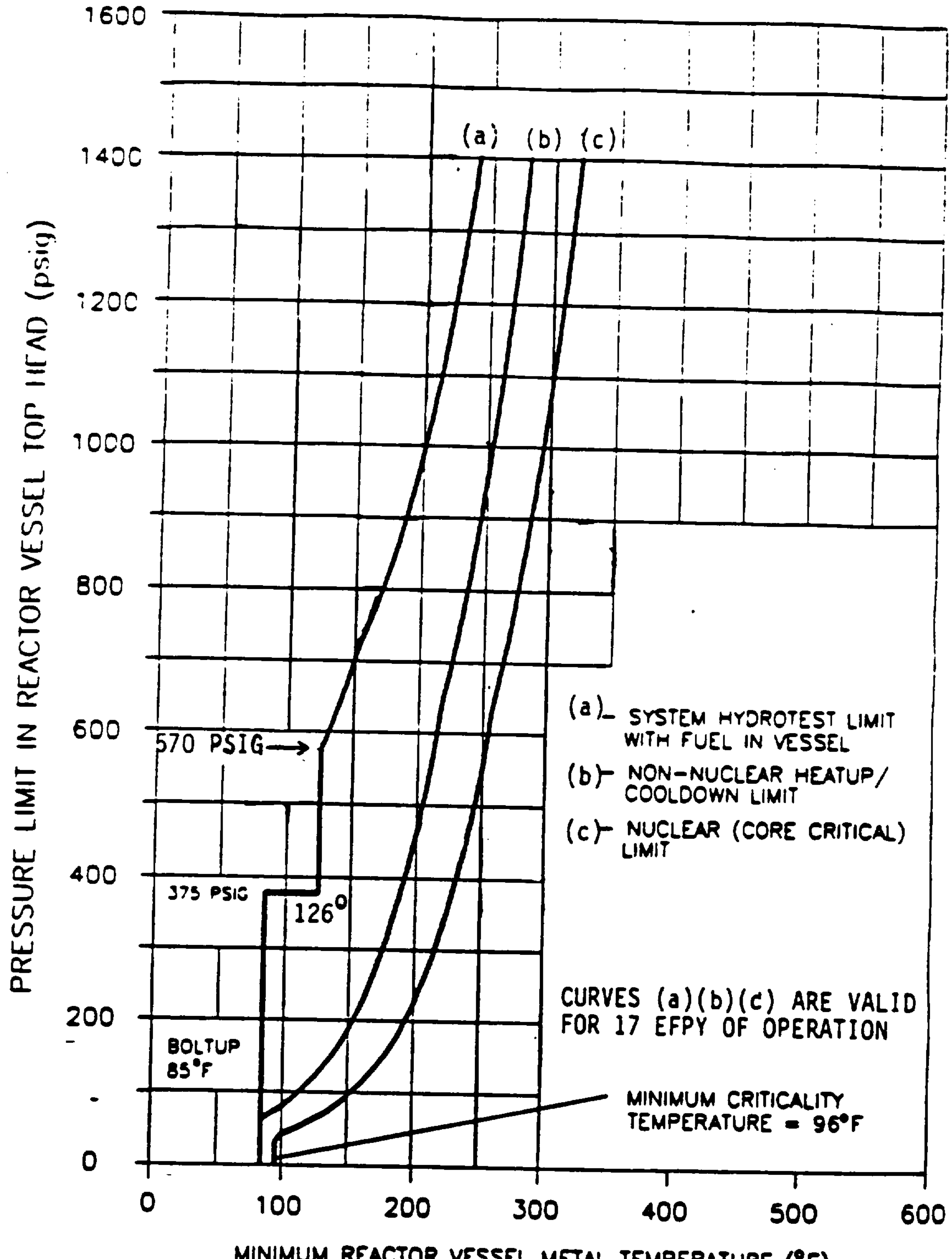
The requirements of Specification 3.3.F.2 for partial loop operation in which the idle loop is isolated, preclude the inadvertent startup of a recirculation pump with a cold leg thus avoiding any reactivity addition transient or reactor vessel nozzle thermal stress concerns.

Specifications 3.3.F.4 and 3.3.F.6 assure that an adequate flow path exists from the annular space, between the pressure vessel wall and the core shroud, to the core region. This provides sufficient hydraulic communication between these areas, thus assuring that reactor water instrument readings are indicative of the level in the core region. For the bounding loss of feedwater transient[(2)], a single fully open recirculation loop transfers coolant from the annulus to the core region at approximately five times the boiloff rate with no forced circulation[(3)]. With the reactor vessel flooded to a level above 185 inches TAF or when the steam separator and dryer are removed, the core region and all recirculation loops can therefore be isolated. When the steam separator and dryer are removed, safety limit 2.1.D ensures water level is maintained above the core shroud.

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) SASR 90-89 (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 LIMITS



4.3 REACTOR COOLANT

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 Figure 3.3.1. 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
4	1221 ± 12
1	1230 ± 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. Pri_cy Coolant System Pressure Isol_ion Valves 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.

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 Figure 3.3.1 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

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(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
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 151

TO PROVISIONAL 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 January 11, 1991, as supplemented March 12, 1991, GPU Nuclear Corporation (GPUN, the licensee) requested permission to revise the pressure/temperature (P/T) limits in the Oyster Creek Technical Specifications (TS) Section 3.3.A. The proposed P/T limits were requested for 17 effective full power years (EFPY). GPUN also requested to revise TS Section 3.3.B to provide a new reactor vessel temperature limit for full tensioning of the reactor vessel closure head studs. Current Oyster Creek TS show P/T operating curves in three different figures. This request provides all three P/T curves in one figure (Figure 3.3.1). To reflect this arrangement, editorial changes are made in TS Section 3.3.A and its Bases, and TS Section 4.3.A and its Bases. We find these editorial changes acceptable.

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

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

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

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

2.0 EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Oyster Creek reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff determined that the material with the highest ART at 17 EFPY was the intermediate shell course plate G-8-6 with 0.20% copper (Cu), 0.51% nickel (Ni), and an initial RT_{ndt} of 31°F.

The licensee has removed one surveillance capsule, 210, from Oyster Creek. The results from capsule 210 were published in Battelle Columbus Laboratories Report BCL-382-85-1 (R1). The surveillance capsule contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, plate G-8-6, the staff calculated the ART to be 137.4°F at 1/4T (T = reactor vessel beltline thickness) and 116.1°F for 3/4T at 17 EFPY. The staff used a neutron fluence of 1.25E18 n/cm² at 1/4T and 5.5E17 n/cm² at 3/4T. The ART was determined using Section 1 of RG 1.99, Rev. 2, because only one surveillance capsule has been removed from the reactor vessel.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 137.5°F at 1/4T for the same limiting plate metal. Substituting the ART of 137.5°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for hydrotest and cooldown meet the SRP 5.3.2; however, the licensee's heatup limits at 1000 psi is about 19°F less conservative than the staff's calculation.

In the normal BWR operation, the reactor coolant system (RCS) is heated based on the thermodynamic saturated steam line which is much more conservative than the P/T limits. Based on this consideration, the staff believes that the 19°F difference is acceptable; therefore, the proposed P/T limits for heatup is also acceptable.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Paragraph IV.A.3 of Appendix G states "an exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the pre-service system hydrostatic test pressure. In this case the minimum permissible temperature is 60°F (33°C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload." Based on the flange reference temperature of 25°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the limiting (lowest) Charpy USE at end of life be above 50 ft-lb. The plate, G-8-7, has the lowest unirradiated USE of 78 ft-lb. The staff calculated that the EOL USE at 1/4T for this material will be 61.2 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 17 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The proposed limits also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Rev. 2 to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Oyster Creek Technical Specifications.

3.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32 and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on April 11, 1991 (56 FR 14715). Accordingly, based upon the environmental assessment, we have determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

4.0 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 nor to the health and safety of the public.

5.0 REFERENCES

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

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

3. January 11, 1991, Letter from J. J. Barton (GPUN) to USNRC Document Control Desk, Subject: Oyster Creek Nuclear Generating Station, Technical Specification Change Request No. 194.

4. M. P. Manahan, et. al., "Final Report on Examination, Testing, and Evaluation of Specimens from the 210-Degree Irradiated Pressure Vessel Surveillance Capsule for the Oyster Creek Nuclear Generating Station," BCL-382-85-1 (Revision 1), October 18, 1985.

Principal Contributor: J. Tsao

Date: April 11, 1991

UNITED STATES NUCLEAR REGULATORY COMMISSION

GPU NUCLEAR CORPORATION

DOCKET NO. 50-219

NOTICE OF ISSUANCE OF AMENDMENT TO

PROVISIONAL OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 151 to Provisional Operating License No. DPR-16 issued to GPU Nuclear Corporation (the licensee), which revised the Technical Specifications for operation of the Oyster Creek Nuclear Generating Station located in Ocean County, New Jersey. The amendment is effective as of the date of issuance.

The amendment revised Technical Specification (TS) 3.3 of the Oyster Creek Nuclear Generating Station TS. Specifically, the amendment revised TS Section 3.3.A, pressure/temperature (P/T) limits of the reactor coolant system for operation up to 17 effective full power years. The amendment also revised TS Section 3.3.B to provide a new reactor vessel temperature limit for full tensioning of the reactor vessel closure head studs.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on February 11, 1991 (56 FR 5431). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment.

For further details with respect to the action see (1) the application for amendment dated January 11, 1991, as supplemented March 12, 1991, (2) Amendment No. 151 to License No. DPR-16, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, N.W., Washington, D.C. and at the local public document room, Ocean County Library, Reference Department, 101 Washington Street, Toms River, New Jersey 08753. A copy of items (2), (3) and (4) may be obtained upon requested addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects - I/II.

Dated at Rockville, Maryland this 11th day of April 1991.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation