March 6, 1991

Docket No. 50-219

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

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Dear Mr. Barton:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 75536)

The Commission has issued the enclosed Amendment No.150 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated December 18, 1989, as supplemented April 30, October 16, and November 16, 1990.

The amendment revises the Technical Specifications to permit the removal of seven main steam safety valves with the two highest setpoints.

A copy of the related Safety Evaluation is also 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. 150to DPR-16
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

OFC	:PDI-4:LA	:PDI-4:PM	:PDI-4:D	:OGC SLU	:
NAME	:SNorris	:ADromerick:cn	:JFStolz	: S.U.	:
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Mr. J. J. Barton

Oyster Creek Nuclear
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cc:

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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. 150
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 December 18, 1989, as supplemented April 30, October 16, and November 16, 1990, 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. , 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: March 6, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 150

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
2.3-2	2.3-2
2.3-6	2.3-6
4.3-1	4.3-1

FUNCTION	LIMITING SAFETY SYSTEM SETTINGS
B. Neutron Flux, Control Rod Block	The Rod Block setting shall be $S \leq [(0.90 \times 10^{-6})\ W + 53.1]\ \left[\frac{FRP}{MFLPD}\right]$ with a maximum setpoint of 108% for core flow equal to 61 x 10^6 lb/hr and greater. The definitions of S, W, FRP and MFLPD used above for the APRM scram trip apply. The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than 1.0, in which case the actual operating value will be used. This adjustment may be accomplished by increasing the APRM gain and thus reducing the flow referenced APRM rod block curve by the reciprocal of the APRM gain change.
C. Reactor High, Pressure, Scram	≤1060 psig
D. Reactor High Pressure, Relief Valves Initiation	2 @ ≤ 1070 psig 3 @ ≤ 1090 psig
E. Reactor High Pressure, Isolation Condenser Initiation	≤1060 psig with time delay ≤3 seconds
F. Reactor High Pressure, Safety Valve Initiation	4 @ 1212 psig ±12 psi 4 @ 1221 psig ±12 psi 1 @ 1230 psig ±12 psi
G. Low Pressure Main Steam Line, MSIV Closure	≥825 psig (initiated in IRM range 10)
H. Main Steam Line Isolation Valve Closure, Scram	≤10% Valve Closure from full open
I. Reactor Low Water Level, Scram	≥11'5" above the top of the active fuel as indicated under normal operating conditions
J. Reactor Low-Low Water Level, Main Steam Line Isolation Valve Closure	≥7'2" above the top of the active fuel as indicated under normal operating conditions

The reactor coolant system safety valves offer yet another protective feature for the reactor coolant system pressure safety limit since these valves are sized assuming no credit for other pressure relieving devices. In compliance with Section I of the ASME Boiler and Pressure Vessel Code, the safety valve must be set to open at a pressure no higher than 103% of design pressure, and they must limit the reactor pressure to no more than 110% of design pressure. The safety valves are sized according to the Code for a condition of main steam isolation valve closure while operating at 1930 MWt, followed by [1] a reactor scram on high neutron flux, [2] failure of recirculation pump trip on high pressure, [3] failure of the turbine bypass valves to open, and [4] failure of the isolation condensers and relief valves to operate. Under these conditions, a total of 9 safety valves are required to turn the pressure transient. The ASME B&PV Code allows a ±1% of working pressure (1250 psig) variation in the lift point of the valves. This variation is recognized in Specification 4.3.

The low pressure isolation of the main stream lines at 825 psig was provided to give protection against fast reactor depressurization and the resulting rapid cool-down of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 825 psig requires that the reactor mode switch be in the STARTUP position and the IRMs be in the range 9, or lower, where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line low pressure isolation and isolation valves closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition the isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure.

The low water level trip setting of 11'5" above the top of the active fuel has been established to assure that the reactor is not oporated at a water level below that for which the fuel cladding integrity safety limit is applicable. With the scram set at this point, the generation of steam, and thun the loss of inventory, is stopped. For example, for a loss of feedwater flow a reactor scram at the value indicated and isolation valve closure at the low-low water level set point results in more than 4 feet of water remaining above the core after isolation (6).

During periods when the reactor is shut down, decay heat is present and adequate water level must be maintained to provide core cooling. Thus, the low-low level trip point of 7'2" above the core is provided to actuate the core spray system (when the core spray system is required as identified in Section 3.4) to provide cooling water should the level drop to this point.*

The turbine stop valve(s) scram is provided to anticipate the pressure, neutron flux, and heat flux increase caused by the rapid closure of the turbine stop valve(s) and failure of the turbine bypass system.

The generator load rejection scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control

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 speciments 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 of Figures 3.3.1(a), (b) and (c). 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 150

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 December 18, 1989, GPU Nuclear Corporation (GPUN/the licensee) requested changes to the Oyster Creek Technical Specifications (TS). GPUN submitted additional information by letters dated April 30, October 16, and November 16, 1990. The proposed TS will permit removal of seven main steam safety valves (SVs). The current plant configuration includes 16 SVs and 5 electromatic relief valves (EMRVs). The original design for Oyster Creek overpressure protection was based on ASME Section I, 1962 Edition. The ASME Section I deals specifically with power boilers and not nuclear reactors. (In 1962, there was no ASME Section III which deals with nuclear reactors.) This edition was interpreted to require that the safety valve capacity be such that all potentially generated steam by the boiler (reactor vessel) be discharged without credit for fuel stoppage. The current versions of the ASME code allow (Sections I and III) credit for operating or safety controls in the boiler or nuclear reactor. GPUN proposes to take credit for the high flux scram and plans to remove 7 out of 16 safety valves provided at Oyster Creek. The licensee wishes to remove the valves to 1) reduce worker radiation exposure associated with valve maintenance; 2) reduce the likelihood of inadvertent valve opening or leakage; and 3) to reduce maintenance costs.

2.0 EVALUATION

At Oyster Creek, there are two emergency isolation condensers which are started automatically on reactor high pressure. There are five electromatic relief valves (EMRVs). The turbine bypass capability is 40%. In addition to the above, there are 16 safety valves provided. The combined relief and safety valve capacity is about $13x10^6$ #/hr which is approximately 180% of the total steam flow. The probability of lifting SVs is low due to the availability of the isolation condensers and the five EMRVs. The EMRVs discharge to the torus (suppression pool) and the SVs discharge to the drywell.

GPUN performed an overpressure protection analyses to verify that nine SVs are sufficient to meet the acceptance criteria for overpressure protection. The impact on ATWS response was also examined.

2.1 Overpressure Protection-Main Steam Isolation Valve (MSIV) Closure With High Flux Scram , No Recirculation Pump Trip (RPT) (Nine SVs And No RPT)

The RETRAN-02 Mod 4 code was used for this analysis. GPUN has based the analysis on the initiation of a reactor scram by the high-neutron flux signal which is the second safety grade scram signal from the reactor protection system following MSIV closure. It was assumed that the position switches of the MSIVs failed to scram the reactor. The analysis took credit for only nine SVs. No credit was given to the five EMRVs, the RPT, or the isolation condensers. The results of these analyses demonstrate that the maximum pressure will remain below the 1375 psig limit. For the most severe transient (closure of all MSIVs with a high neutron flux scram at 120%), the maximum vessel pressure is 1361 psig when only nine SVs are assumed to operate in the safety mode. Since the calculated peak pressure 1361 psig is within the acceptance criterion of 1375 psig, the overpressure protection analysis is acceptable.

2.2 Anticipated Transient Without Scram (ATWS)-MSIV Closure ATWS (8 Safety Valves) With RPT

A MSIV closure ATWS event with RPT was evaluated to assess the effect of removal of safety valves on transient response. The transient was analyzed with eight safety valves, five EMRVs and with RPT. The peak calculated pressure was 1282 psig which is well within the ASME service level C overpressure limit (1500 psig) typically used as a guideline for ATWS analyses.

Since there are only nine safety valves instead of 16, the peak reactor pressure during the transient is increased. This increased pressure will cause a slight increase in the amount of steam discharged to the pool initially through the EMRVs. The additional heat input is small because it is due primarily to the higher initial pressure spike. With the pool cooling system in operation (as procedures require during an ATWS) the containment pressure and temperature parameters are not expected to change significantly relative to the previous plant configuration.

GPUN performed overpressurization analyses consistent with SRP 5.2.2 to support its proposal to removal seven safety valves at Oyster Creek. The results were consistent with the NRC staff's acceptance criteria that pressure not exceed 110% of design. The licensee also discussed the impact on ATWS response indicating no significant impacts. The GPUN proposal to remove seven safety valves from the current 16 safety valves is therefore acceptable. The changes proposed in Technical Specifications 2.3.F and 4.3.E reducing the number of safety valves and the bases are also acceptable.

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 February 26, 1991 (56 FR 7883). Accordingly, based upon the environmental assessment, we have determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

4.0 CONCLUSION

The staff 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, and (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.

Principal Contributor: G. Thomas

Dated: March 6, 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. 150 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 modified the Technical Specifications to permit the removal of seven main steam safety valves with the two highest setpoints.

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 1, 1990 (55 FR 3502). 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 December 18, 1989, as supplemented April 30, October 16, and November 16, 1990, (2) Amendment No. 150 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 request 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 6th day of March 1991.

FOR THE NUCLEAR REGULATORY COMMISSION

Alexander W. Dromerick

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