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

May 22, 1985

Amdt. 82 to DPR-16

Docket No. 50-219
LS05-85-05-024

Mr. P. B. Fiedler
Vice President and Director
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Fiedler:

SUBJECT: INSERVICE INSPECTION AND TESTING

Re: Oyster Creek Nuclear Generating Station

The Commission has issued the enclosed Amendment No.82 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station. This amendment is in response to your application dated June 8, 1984, which supersedes your application dated December 11, 1979.

The amendment authorizes administrative revisions to Inservice Inspection (ISI) and Inservice Testing (IST) requirements in Section 4.3, Reactor Coolant, of the Oyster Creek Appendix A Technical Specifications (TS). The Bases for Section 4.3 refers to a table and to figures in the Oyster Creek Facility Description and Safety Analysis Report. You are requested to revise the Bases for TS Section 4.3 to refer to the appropriate sections in the recently issued Oyster Creek updated Final Safety Analysis Report.

You revised your June 8, 1984 submittal by letter dated February 11, 1985, by proposing additional changes to Section 4.3 of the T.S. The additional proposed changes to the ISI/IST requirements in Section 4.3 of the TS that were in the February 11, 1985 submittal and not in the June 8, 1984 submittal will be the subject of a future action by the staff.

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on February 27, 1985, (50 FR 7988). No public comments or requests for hearing were received.

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A copy of our related Safety Evaluation is also enclosed. A notice of issuance pertaining to this action will appear in the Commission's monthly notice publication in the Federal Register.

Sincerely,

Original signed by:
John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosures:
1. Amendment No. 82 to License No. DPR-16
2. Safety Evaluation

cc w/enclosures:
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

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 82
License No. DPR-16

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

   A. The application for amendment by GPU Nuclear Corporation and Jersey Central Power and Light Company (the licensees) dated June 8, 1984, which supersedes the application dated December 11, 1979, 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. 82, 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 its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 22, 1985.

ATTACHMENT TO LICENSE AMENDMENT NO. 82

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
4.3-1	4.3-1
4.3-1a	-
4.3-2	4.3-2
4.3-3	4.3-3
4.3-4	-
4.3-5	-
4.3-6	-
4.3-7	-
4.3-8	-
4.3-9	4.3-4

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. Neutron flux monitors shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The monitors shall be removed and tested at the first refueling outage to experimentally verify the calculated values of integrated neutron flux that are used to determine the NDTT from Figure 3.3.1.

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. 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 bench checked for proper set point. A minimum of 5 of the valves shall be bench checked or replaced with a bench checked valve each refueling outage such that all valves are checked in three successive refueling outages, to ensure set points are as follows:

Number of Valves	Set Point (psig)
4	1212 ± 12
4	1221 ± 12
4	1230 ± 12
4	1239 ± 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. Primary Coolant System Pressure Isolation 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 on 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, and prior to returning the valve to service after maintenance, repair or replacement work is performed.

(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.

Bases:

Numerous data are available relating integrated flux and the change in Nil-Ductility Transition Temperature (NDTT) in various steels. The base metal has been demonstrated to be relatively insensitive to neutron irradiation (see expected NDT changes in FDSAR Table IV-1-1, and Figures IV-2-9 and IV-2-10). The most conservative data has been used in Specification 3.3. The integrated flux at the vessel wall is calculated from core physics data and will be measured using flux monitors installed inside the vessel. The measurements of the neutron flux at the vessel wall will be used to check and if necessary correct, the calculated data to determine an accurate flux. From this a conservative NDT temperature can be determined. Since no shift will occur until an integrated flux of 10^{17} nvt is reached, the confirmation can be made long before an NDTT shift would occur.

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 the existence of ASME Section XI. For this reason, the degree of access required by ASME Section XI is not generally available and will be addressed as "requests for relief" in accordance with 10 CFR 50.55a(g).

Experience in safety valve operation shows that a check of approximately 1/3 of the safety valves per year is adequate to detect failures or deterioration. The tolerance value is specified in Section I of the ASME Code at +1% of design pressure. An analysis has been performed which shows that with all safety valves set 12 psig higher the safety limit of 1375 psig is not exceeded.

Conductivity instruments continuously monitor the reactor coolant. Experience indicates that a check of the conductivity instrumentation at least every 72 hours is adequate to ensure accurate readings. The reactor water sample will also be used to determine the chloride ion content to assure that the limits of 3.3.E are not exceeded. The chloride ion content will not change rapidly over a period of several days; therefore, the sampling frequency is adequate.

TABLE 4.3.1

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

System	Valve No.	Maximum (a) Allowable Leakage
Core Spray System 1	NZ02A	5.0 GPM
	NZ02C	5.0 GPM
Core Spray System 2	NZ02B	5.0 GPM
	NZ02D	5.0 GPM

Footnote:

(a) 1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

5. Test differential pressure shall not be less than 150 psid.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 82 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 June 8, 1984 which superseded the December 11, 1979, request, GPU Nuclear Corporation (GPUN) (the licensee) requested an amendment to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station. This amendment would authorize changes to Section 4.3, Reactor Coolant, of the Appendix A Technical Specifications. The changes revise the inservice inspection and inservice testing requirements in Section 4.3.

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on February 27, 1985 (50 FR 7988). No public comments or requests for hearing were received.

2.0 DISCUSSION AND EVALUATION

On February 27, 1976, the Nuclear Regulatory Commission revised the inservice inspection and testing requirements for ASME Code Class 1, 2, and 3 components for nuclear power plants in 10 CFR 50.55a. The revised regulations require inservice inspection and testing set forth in Section XI of the ASME Boiler and Pressure Vessel Code and Addenda. A review by the Commission of the 1974 edition ASME Section XI indicated that conflicts may occur between the ASME code requirements and the plant Technical Specifications (TS). To avoid such conflicts, the Commission requested that the licensee, in accordance with 50.55a(g)(5)(ii), apply for an amendment to the plant TS to replace such conflicting TS with a reference to 10 CFR 50.55a. The licensee proposed by an amendment request dated June 8, 1984 to incorporate the requirements of the revised regulations on inservice inspection and testing in the plant TS.

The licensee previously, by amendment request dated December 11, 1979, proposed to delete nondestructive examination requirements for the reactor coolant system from Section 4.3 of the TS because that requirement was contained in the Oyster Creek Inservice Inspection Program for the second

10-year interval and also proposed to renumber TS pages and a table in Section 4.3 as needed to accommodate the proposed changes. The requirements requested to be deleted are existing TS 4.3.B and Table 4.3.1. This request was included in the letter dated June 8, 1984.

The staff has reviewed the changes proposed by the licensee in its letter dated June 8, 1984. The proposed changes, to incorporate the requirements of the revised 10 CFR 50.55a and to avoid conflicts between the ASME Section XI requirements and the plant TS, add a revised TS 4.3.B and 4.3.C. These additional TS are consistent with the guidance provided by the staff in the letter dated April 26, 1976, to the licensee. Therefore, the staff concludes that these proposed changes to the TS are acceptable.

The proposed changes, which would delete detailed requirements in the TS that are in the Oyster Creek Inservice Inspection Program, delete existing TS 4.3.B and Table 4.3.1. The existing TS 4.3.B requires that nondestructive examinations shall be made in accordance with Table 4.3.1 and that any indication of a defect shall be investigated and evaluated. These requirements are in the Oyster Creek Inservice Inspection Program. This program was reviewed and approved in the Safety Evaluation dated June 28, 1983. Therefore, the staff concludes that these proposed changes are acceptable.

Therefore, based on the above, the staff concludes that the changes to Section 4.3 of the Appendix A TS proposed in the licensee's letter dated June 8, 1984, are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this 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 this amendment.

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

5.0 ACKNOWLEDGEMENT

This evaluation was prepared by G. Johnson, J. Page, and J. Donohew.

Dated: May 22, 1985