50-219

September 27, 1994

Mr. John J. Barton
Vice President and Director
GPU Nuclear Corporation
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, NJ 08731

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SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M89348)

Dear Mr. Barton:

The Commission has issued the enclosed Amendment No. 170 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated April 19, 1994.

The amendment updates and clarifies Technical Specification (TS) 3.4.B.1 to be consistent with TSs 1.39 and 4.3.D. It addresses electromatic relief valve operability/bypassing during system pressure testing, including system leakage and hydrostatic test, with the reactor vessel solid, core not critical, and core reactivity limits satisfied.

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

Sincerely,

Original signed by R. Hernan
for

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. 170 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
GPU Nuclear Corporation
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Sincerely,

Ronald W. Hernan for

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

Docket No. 50-219

Enclosures: 1. Amendment No. 170 to DPR-16
2. Safety Evaluation

cc w/encls: See next page

Mr. John J. Barton
GPU Nuclear Corporation

Oyster Creek Nuclear
Generating Station

cc:

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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. 170
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 April 19, 1994, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No.170 , 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 60 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: September 27, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 170

FACILITY OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

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

Remove	Insert
Page 3.4-4	Page 3.4-4

automatic depressurization function) may be inoperable or bypassed during Reactor Vessel Pressure Testing consistent with Specifications 1.39 and 3.3.A.(i).

2. If at any time there are only four operable electromatic relief valves, the reactor may remain in operation for a period not to exceed 3 days provided the motor operated isolation and condensate makeup valves in both isolation condensers are verified daily to be operable.

3. If Specifications 3.4.B.1 and 3.4.B.2 are not met; reactor pressure shall be reduced to 110 psig or less, within 24 hours.

4. The time delay set point for initiation after coincidence of low-low-low reactor water level and high drywell pressure shall be set not to exceed two minutes.

C. Containment Spray System and Emergency Service Water System

1. The containment spray system and the emergency service water system shall be operable at all times with irradiated fuel in the reactor vessel, except as specified in Specifications 3.4.C.3, 3.4.C.4, 3.4.C.6 and 3.4.C.8.

2. The absorption chamber water volume shall not be less than 82,000 ft^3 in order for the containment spray and emergency service water system to be considered operable.

3. If one emergency service water system loop becomes inoperable, its associated containment spray system loop shall be considered inoperable. If one containment spray system loop and/or its associated emergency service water system loop becomes inoperable during the run mode, the reactor may remain in operation for a period not to exceed 7 days provided the remaining containment spray system loop and its associated emergency service water system loop each have no inoperable components and are verified daily to be operable.

4. If a pump in the containment spray system or emergency service water system becomes inoperable, the reactor may remain in operation for a period not to exceed 15 days provided the other similar pump is verified daily to be operable. A maximum of two pumps may be inoperable provided the two pumps are not in the same loop. If more than two pumps become inoperable, the limits of Specification 3.4.C.3 shall apply.

5. During the period when one diesel is inoperable, the containment spray loop and emergency service water system loop connected to the operable diesel shall have no inoperable components.

6. If primary containment integrity is not required (see Specification 3.5.A), the containment spray system may be made inoperable.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 170

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

The GPU Nuclear Corporation (the licensee) letter of April 19, 1994 request for a Technical Specification (TS) change (Ref. 1) is to update and clarify TS 3.4.B.1 to be consistent with TSs 1.39 and 4.3.D. The Technical Specification Request (TSCR) deletes reference to the ASME Code Section XI, IS-500 ten-year hydrotest inspection interval and replaces this with references to (1) the Technical Specification 1.39 definition for Reactor Vessel Pressure Testing and (2) the Technical Specification 3.3.a(i) Reactor Vessel Pressure Testing limits (P/T 250°F maximum test temperature).

The TSCR clarifies that the five electromatic relief valves (EMRV) pressure relief function may be inoperable or bypassed during system pressure testing required by ASME Code Section XI, ARTICLE IWA-5000, including system leakage and hydrostatic test, with reactor vessel completely solid, core not critical and Technical Specification 3.2.A (core reactivity limits) satisfied.

2.0 DISCUSSION

Technical Specification 4.3.D addresses ASME Code Section XI Article 5000 requirements for pressure leak testing of the reactor pressure boundary following refueling outages or major repairs to the reactor coolant system (RCS). The licensee states that pressure leak testing requires a pressure of 1055 +10/-0 psig, a requirement inconsistent with the EMRV setpoint of 1060 psig (Ref. 1). The inconsistency is similar to one addressed by the staff in Amendment 44 (Ref. 2) in which it stated:

> The licensee has also proposed to allow the pressure relief function of the electromatic relief valves to be inoperable or bypassed (the ADS [automatic depressurization system] function of the valves would be maintained) during the system hydrostatic pressure test required by AMSE [sic] Code Section XI, IS-500 at or near the end of each ten year inspection interval. This allowance is necessary since the hydrostatic test pressure is above the setpoint of the relief valves.

> Even though the pressure relief function of the electromatic relief valves is bypassed, over pressure protection would continue to be provided by the 16 [code] safety valves. Elimination of this relief function does not affect the reactor safety analyses, since credit was not taken for the relief function. Therefore, we find the modification acceptable.

The staff has been informed that the statement regarding the safety valves was incorrect for the 10 year hydro test since the valves are gagged to prevent weeping (Refs. 3 and 4). Therefore, in an SE issued August 29, 1994, the staff concluded that it is acceptable to gag the safety valves during a hydrostatic system test. However, the EMRVs are configured to provide overpressure protection for that test. In the leak testing being addressed herein, the safety valves are not gagged and the mistake has no impact.

The staff later, in Amendment 150 (Ref. 5), determined that the 16 code safety valves could be reduced to nine on the basis of analyses that the staff judges to bound shutdown-related conditions appropriate to the Reference 1 request. (The Standard Review Plan does not address operation of boiling water reactors under the shutdown conditions of concern here.) Although the licensee concludes in its Reference 1 request that only non-critical conditions are appropriate, the staff considered inadvertent criticality in addition to non-critical conditions. The staff concludes that an inadvertent criticality is unlikely. The licensee assures that a withdrawn rod will not result in criticality prior to the pressurization test (Ref. 6). The moderator temperature coefficient will be negative under the test conditions (Ref. 6) and heatup will not induce criticality. Even if a criticality were to occur, the doppler coefficient will inhibit the power increase and any void formation would reduce reactivity. The staff is satisfied that the combination of unlikely criticality, limiting characteristics, and design basis analyses means that criticality need not be considered further.

3.0 EVALUATION

Bypassing the EMRV actuation (setpoint of 1060 psig) will prevent operation of the automatic pressure relief function, but will have no effect on automatic or operator-actuated depressurization capability provided by the EMRVs. Reactor coolant system overpressure protection will continue to be provided by the remaining nine code safety valves, four of which have a setpoint of 1212 psig, rather than the EMRV setpoint of 1060 psig (Refs. 1 and 6).

The test process is to (Ref. 7):

- Maintain temperature between 215 °F and 225 °F by operation of reactor recirculation pumps (Sections 4.1.1, 6.22.1).
- Maintain pressure between 1055 psig and 1065 psig by varying letdown through the reactor water cleanup system and by controlling flow to the control rod drives (CRDs) (Section 4.1.1).

- Use one CRD pump (Section 4.1.1).

Precautions and limitations include (Ref. 7):

- Limit pressurization rate to 150 psig/min. (Sections 4.2.1, 6.23)
- Maintain temperature ≥ 215 °F while at 1055 - 1065 psig and < 240 °F as read at the suction of an operating recirculation pump (Sections 4.2.3, 4.2.5)
- Limit reactor coolant temperature increases so that the temperature difference between reactor head metal and recirculation pump suction does not exceed 50 °F (Sections 4.2.6 and 6.22.1).

Other significant Reference 7 Sections include:

6.3.2 "Run at least one recirc. pump from the time heat-up to test temperature is commenced until pressure is reduced to atmospheric."

6.22.2 "Soak reactor vessel at test temperature (215-225°F) for one hour prior to initiating pressurization."

The test procedure is consistent with Oyster Creek pressure/temperature limits specified in TS Figures 3.3.1.

The staff notes that the test procedure also addresses such potential concerns as:

- The only valid measurement of RCS water temperature is obtained via thermocouples located at the recirculation pump inlet. A meaningful indication of RCS temperature is obtained only if there is significant flow in the recirculation pump piping.
- Thermal stratification is a concern unless there is significant flow and RV water mixing. This is particularly true for the RV lower plenum and head, where cold water from the CRD could accumulate, with obvious implications to the RV wall temperature requirements of Figures 3.3.1 and to the temperature that would be measured in the pump piping.
- There is a time lag between heating of RV water and the response of RV metal temperature. Failure of the RV metal temperature to be adequately raised would result in a violation of TS Figures 3.3.1.

4.0 TECHNICAL SPECIFICATION CHANGE

The change is as follows where the new wording is in **bold type:**

3.4.B. Automatic Depressurization System

1. Five electromatic relief valves of the automatic depressurization system shall be operable when the reactor water temperature is greater than 212 °F and pressurized above 110 psig, except as specified in 3.4.B.2. The automatic pressure relief function of these valves (but not the automatic depressurization function) may be inoperable or bypassed during **Reactor Vessel Pressure Testing consistent with Specifications 1.39 and 3.3.A.(i).**

Based on the staff evaluation above, the staff concludes that the proposed Technical Specification change is acceptable.

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (59 FR 27056). 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.

7.0 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.

8.0 REFERENCES

1. Barton, J. J., "Oyster Creek Nuclear Generating Station, Docket No. 50-219, Facility Operating License No. DPR-16, Technical Specification Change Request No. 213, Reactor Pressure Vessel Testing - EMRV (electromatic relief valve) Bypass," Letter to NRC, C321-93-2376, April 19, 1994.

2. "Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 44 to Provisional Operating License No. DPR-16, Jersey Central Power & Light Company, Oyster Creek Nuclear Generating Station, Docket No. 50-219," NRC, January 4, 1980.

3. Barton, J. J., "Oyster Creek Nuclear Generating Station, Docket No. 50-219, Facility Operating License No. DPR-16, ASME Code System Hydrostatic Pressure Testing," Letter to NRC from Vice President and Director, Oyster Creek, C321-94-2081, June 3, 1994.

4. Barton, J. J., "Oyster Creek Nuclear Generating Station, Docket No. 50-219, Ten Year Hydrostatic Test," Letter to NRC from Vice President and Director, Oyster Creek, C321-94-2121, August 1, 1994.

5. "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," NRC, March 6, 1991.

6. Zak, Ron (GPU Nuclear), Telephone call with Warren Lyon and Ronald Hernan (NRC) to confirm staff understanding of licensee operations, August 16, 1994.

7. "Nuclear Steam Supply System (NSSS) Leak Test," Oyster Creek Nuclear Generating Station Procedure Number 602.4.001 Rev. 19, pp 1 - 25, February 19, 1993. Received from Ron Zak (GPU Nuclear) as the latest version of the applicable procedure, August 16, 1994.

Principal Contributor: Warren Lyons

Date: September 27, 1994