October 27, 1986

Docket No. 50-219

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: AUTOMATIC DEPRESSURIZATION SYSTEM OPERABLE SURVEILLANCE REQUIREMENTS (TSCR 140, TAC 62978)

Re: Oyster Creek Nuclear Generating Station

The Commission has issued the enclosed Amendment No. 109 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station. This amendment is in response to your application dated September 11, 1986.

This amendment authorizes two changes to Section 4.4, Emergency Cooling, of the Appendix A Technical Specifications (TS), which lists the surveillance requirements and the frequency of surveillance for the reactor emergency cooling systems. This amendment changes (1) the stated frequency and pressure conditions for the Automatic Depressurization System (ADS) valve operability test in Item 4.4.B.1 of Section 4.4, Emergency Cooling, to after each refueling outage and at system operating pressure prior to exceeding 5 percent power and (2) revises the Bases for this item in Section 4.4. This change was to clarify the surveillance requirements of ADS valve operability in the TS. This change to clarify the TS was needed prior to the restart from the current Cycle 11 Refueling outage.

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

Sincerely,
Original signed by

Jack N. Donohew, Jr., Project Manager
BWR Project Directorate #1
Division of BWR Licensing

Enclosures:
1. Amendment No. 109 to
   License No. DPR-16
2. Safety Evaluation

cc w/enclosures:
See next page

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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. 109
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 September 11, 1986, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 109, 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

Jack N. Donohew Jr

Jack N. Donohew, Jr., Project Manager
BWR Project Directorate #1
Division of BWR Licensing

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: October 27, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 109

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

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

REMOVE	INSERT
4.4-1	4.4-1
4.4-2	4.4-2

4.4 EMERGENCY COOLING

Applicability: Applies to surveillance requirements for the emergency cooling systems.

Objective: To verify the operability of the emergency cooling systems.

Specification: Surveillance of the emergency cooling systems shall be performed as follows:

Item	Frequency
A. Core Spray System	
1. Pump Operability	Once/month. Also after major maintenance and prior to startup following a refueling outage.
2. Motor operated valve operability	Once/month
3. Automatic actuation test	Every three months
4. Pump compartment water-tight doors closed	Once/week and after each entry
5. Core spray header ΔP instrumentation	
check	Once/day
calibrate	Once/3 months
Test	Once/3 months
B. Automatic Depressurization	
1. Valve operability	Following a refueling outage*
2. Automatic actuation test	Every refueling outage
C. Containment Cooling System	
1. Pump Operability	Once/month. Also after major maintenance and prior to startup following a refueling outage

*Valve operability shall be demonstrated at system operating pressure prior to exceeding 5 percent power.

Item	Frequency
2. Automatic actuation test	Every 3 months
3. Pump compartment water-tight doors closed	Once/week and after each entry
D. Emergency Service Water System	
1. Pump Operability	Once/month. Also after major maintenance and prior to startup following a refueling outage.
2. Automatic actuation test	Every 3 months
E. Control Rod Drive Hydraulic System	
1. Pump Operability	Once/month. Also after major maintenance and prior to startup following a refueling outage.
F. Fire Protection System	
1. Pump and Isolation valve operability	Once/month. Also after major maintenance and prior to startup following a refueling outage.

Bases: It is during major maintenance or repair that a system's design intent may be violated accidentally. Therefore, a functional test is required after every major maintenance operation. During an extended outage, such as a refueling outage, major repair and maintenance may be performed on many systems. To be sure that these repairs on other systems do not encroach unintentionally on critical standby cooling systems, they should be given a functional test prior to startup.

Motor operated pumps, valves and other active devices that are normally on standby should be exercised periodically to make sure that they are free to operate. Motors on pumps should operate long enough to approach equilibrium temperature to ensure there is no overheat problem. Whenever practical, valves should be stroked full length to ensure that nothing impedes their motion. Engineering judgment based on experience and availability analyses of the type presented in appendix L of the FDSAR indicates that testing these components more often than once a month over a long period of time does not significantly improve the system reliability. Also, at this frequency of testing wearout should not be a problem through the life of the plant.

During tests of the electromatic relief valves, steam from the reactor vessel will be discharged directly to the absorption chamber pool. Scheduling the tests in conjunction with the refueling outage permits the tests to be run at low power, prior to 5 percent power, enhancing the safety of the plant by assuring EMRV operability before higher power levels are reached.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 109 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 September 11, 1986, GPU Nuclear (the licensee) requested an amendment to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station (Oyster Creek). This amendment would authorize two changes to Section 4.4, Emergency Cooling, of the Appendix A Technical Specification (TS), which lists the surveillance requirements and the frequency of surveillance for the reactor emergency cooling systems. This amendment would change (1) the stated frequency and pressure conditions for the Automatic Depressurization System (ADS) valve operability test in Item 4.4.B.1 of Section 4.4 to after each refueling outage and at system operating pressure prior to exceeding 5 percent power and (2) revise the Bases for Section 4.4. This change was to clarify the surveillance requirements of ADS valve operability in the TS.

2.0 DISCUSSION

The licensee has proposed Technical Specification Change Request (TSCR) No. 140 to clarify the frequency and pressure conditions for testing the ADS valve operability required in the TS. The existing TS 4.4.B.1 on ADS valve operability is confusing. The TS refer to "low pressure" for the tests in the Bases of Section 4.4 but the pressure is not defined and refer to a test every refueling outage but the tests are run as the plant is going from the Refueling Mode into the Run Mode as the plant restarts from the refueling outage. The proposed words clearly state when and at what pressure conditions these tests are conducted.

The ADS consists of five automatically or manually activated electromatic relief valves (EMRVs). The ADS is to (1) depressurize the reactor coolant system (RCS) during a small break LOCA to permit the low pressure core spray system to inject water into the core and (2) provide overpressure protection for anticipated plant transients. The ADS is automatically actuated by high drywell pressure and low-low-low reactor water level. These also are indications of a large break LOCA; however, the large break LOCA will depressurize the RCS by itself and the ADS is not needed.

There are three EMRVs on one steam line and 2 EMRVs on the other steam line from the reactor vessel. The position of the EMRVs is shown in the attached figure from the Oyster Creek Updated Final Safety Analysis Report. The EMRVs blow down to the torus suppression pool in the primary containment and not to the drywell.

Specification 3.4.B.1 states that the EMRVs shall be operable when the reactor water temperature is greater than 212°F and pressurized above 110 psig. The existing specifications could be interpreted to not allow EMRV testing at any steam pressure (steam does not exist below 212°F) and at steam pressures representative of those at which the EMRVs would operate. The EMRVs need a steam pressure above 50 psig to open. Testing the valves at representative operating conditions where they would be expected to operate provides the best assurance that these valves will operate satisfactorily if called upon to depressurize the RCS.

3.0 EVALUATION

The licensee has proposed TSCR 140 to clearly state that the EMRVs may be demonstrated operable at RCS operating pressures prior to exceeding 5 percent power. In addition, in order to remove a source of confusion from the TS the reference to low pressure testing of the EMRVs is proposed to be eliminated from the basis section for Section 4.4 of the TS.

The EMRVs are tested for operability as the plant comes out of every refueling outage when there is essentially no decay heat. In the event of a leak or rupture coincident with the test and the failure of all five EMRVs, the Isolation Condensers can depressurize the RCS since there would be little stored energy or decay heat in the fuel. The depressurization capability of the Isolation Condensers is sufficient for testing the ADS following a refueling outage and as necessary during the operating cycle.

The proposed restriction that valve operability shall be demonstrated prior to exceeding 5 percent power adds a restriction to this surveillance requirement that is not in the existing TS.

The ADS is designed to depressurize the RCS during a small break LOCA to permit the low pressure core spray system to inject water into the core. Testing the EMRVs at system pressure represents normal operating parameters and does not expose the plant to conditions beyond which it is designed to operate. All testing of the EMRVs at Oyster Creek has been at these pressures.

Because of its design, the EMRV cannot be tested below RCS steam pressures of 50 psig. The pressure is, by design, on both the front of the main valve disc which acts to open the valve and on the back of the disc, to close the valve. The unbalanced force to keep the valve closed is the enclosed spring (50 psig). To open the valve, an electrical signal to the solenoid assembly opens a pilot valve to bleed steam off the back of the disc and the higher RCS pressure on the front will open the valve. To close the valve, an electrical signal to the same solenoid assembly closes the pilot valve and steam pressure builds up on the back of the valve disc equal to that on the front and the spring closes the valve. The valve manufacturer recommends testing the valve at the reactor operating pressures for which the valve has been designed.

During the test of each EMRV, the RCS pressure could, if not properly controlled, drop in the RCS because the open EMRV is an open hole on the RCS. A significant drop in pressure would cause voiding and reactivity transients in the RCS which are not desired. The RCS pressure is controlled by the turbine pressure regulator controls.

The pressure regulator controls prevent unnecessary rapid depressurization of the reactor coolant system during the test. Either the mechanical or electrical pressure regulator controls reactor pressure during reactor startup, operation, and shutdown. The mechanical pressure regulator is used during reactor startup and shutdown and the electrical pressure regulator is used at reactor pressures in excess of about 980 psig. The mechanical pressure regulator response to changing pressure conditions is significantly slower than that of the electrical pressure regulator. Therefore, the test at reactor operating pressures which are greater than 980 psig will have the better pressure control regulator (Ref. 2 and 3).

Based on the above, the staff concludes that the licensee's proposed changes to the TS in TSCR 140 are acceptable. The proposed changes to the Bases have been reviewed and found to be appropriate.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component 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 nor environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 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 nor to the health and safety of the public.

6.0 REFERENCES

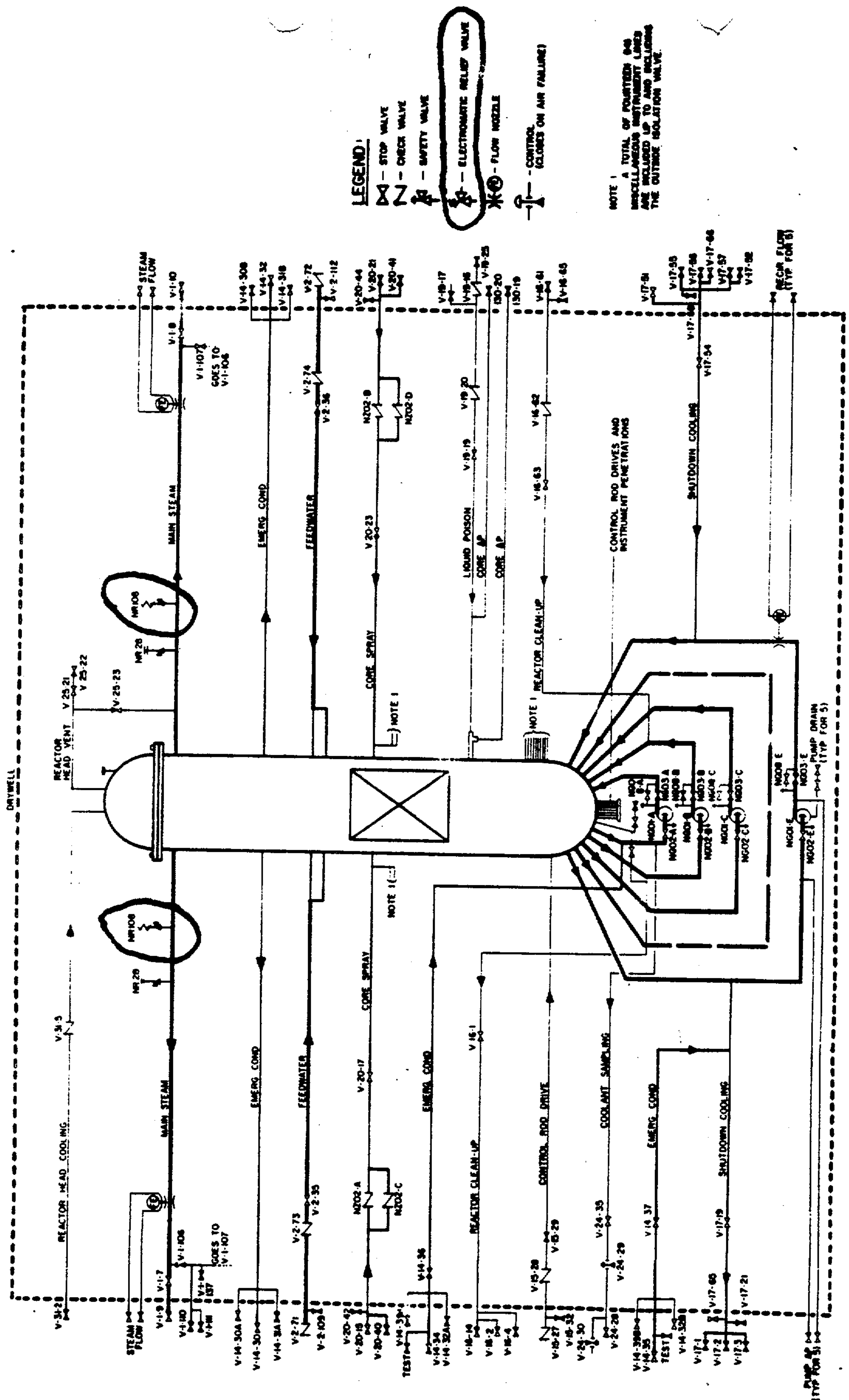
1. Letter from P. B. Fiedler (GPUN) to J. A. Zwolinski (NRC), TSCR No. 140, dated September 11, 1986.

2. Phone conference calls between M. Laggart and J. Kowalski (GPUN) and J. Donohew (NRC) on October 9, 1986.

3. Oyster Creek Nuclear Generating Station, Updated Final Safety Analysis Report, Section 10.2, Turbine Generator.

Principal Contributor: J. Donohew

Dated: October 27, 1986



GPU NUCLEAR CORPORATION OYSTER CREEK NUCLEAR GENERATING STATION UPDATED FINAL SAFETY ANALYSIS REPORT	REACTOR COOLANT SYSTEM BOUNDARIES
REV. 0, 12/84	FIGURE 5.2-1