December 21, 1993

Docket No. 50-219

Mr. John J. Barton
Vice President and Director
GPU Nuclear Corporation
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. M88057)

The Commission has issued the enclosed Amendment No. 167 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated October 18, 1993.

The amendment revises the Technical Specifications to delete requirements to demonstrate by testing, that a redundant system/component is operable when a system/component is declared inoperable. In lieu of testing the redundant system/component to demonstrate its operability, the Technical Specifications are being revised to require an administrative check of plant records to verify operability of the redundant system/component. Conforming changes are made to Definition 1.1 "Operable-Operability."

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:

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. 167 to DPR-16
2. Safety Evaluation

cc w/enclosures:
See next page

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

December 21, 1993

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Vice President and Director
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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. 167
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 October 18, 1993, 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.

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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) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 167, 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: December 21, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 167

FACILITY 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
1.0-1	1.0-1
3.2-3	3.2-3
3.4-1	3.4-1
3.4-4	3.4-4
3.5-6	3.5-6
3.5-7	3.5-7
3.8-1	3.8-1
3.17-1	3.17-1

SECTION I
DEFINITIONS

The following frequently used terms are defined to aid in the uniform interpretation of the specification.

1.1 OPERABLE-OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling of seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

A verification of operability is an administrative check, by examination of appropriate plant records (logs, surveillance test records) to determine that a system, subsystem, train, component or device is not inoperable. Such verification does not preclude the demonstration (testing) of a given system, subsystem, train, component or device to determine operability.

1.2 OPERATING

Operating means that a system or component is performing its required function.

1.3 POWER OPERATION

Power operation is any operation when the reactor is in the startup mode or run mode except when primary containment integrity is not required.

1.4 STARTUP MODE

The reactor is in the startup mode when the reactor mode switch is in the startup mode position. In this mode, the reactor protection system scram trips initiated by condenser low vacuum and main steam line isolation valve closure are bypassed when reactor pressure is less than 600 psig; the low pressure main steamline isolation valve closure is bypassed; the IRM trips for rod block and scram are operable; and the SRM trips for rod block are operable.

1.5 RUN MODE

The reactor is in the run mode when the reactor mode switch is in the run mode position. In this mode, the reactor protection system is energized with APRM protection and the control rod withdrawal interlocks are in service.

1.6 SHUTDOWN CONDITION

The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and there is fuel in the reactor vessel. In this condition, the reactor is subcritical, a control rod block is initiated, all operable control rods are fully inserted, and specification 3. 2-A is met.

2. The standby liquid control solution shall have a Boron-10 isotopic enrichment equal to or greater than 35 atom %, be maintained within the cross-hatched volume-concentration requirement area in Figure 3.2-1 and at a temperature not less than the temperature presented in Figure 3.2-2 at all times when the standby liquid control system is required to be operable.

3.(a) If one standby liquid control system pumping circuit becomes inoperable during the RUN mode and Specification 3.2.A is met, the reactor may remain in operation for a period not to exceed 7 days, provided the pump in the other circuit is verified daily to be operable, otherwise be in the Shutdown condition within 24 hours.

(b) If the solution is outside the cross-hatched volume-concentration area but within the shaded volume-concentration area of Figure 3.2-1, return the solution to the cross-hatched area within 7 days. If, after this time period, the requirement is still not met, submit a report to the NRC within 7 days advising them of plants to return the solution to the cross-hatched volume-concentration area.

(c) If the solution is outside the cross-hatched volume concentration area and outside the shaded volume-concentration area of Figure 3.2-1, return the solution to within the shaded volume-concentration area of Figure 3.2-1 or be in the Shutdown condition within 24 hours.

(d) If the solution temperature is less than the minimum shown in Figure 3.2-2, increase the temperature to greater than the minimum and verify the solution is within the shaded volume-concentration area of Figure 3.2-1 or be in the Shutdown condition within 24 hours.

(e) If the enrichment requirement of 3.2.C.2 is not met:

(1) Return the Boron-10 isotopic enrichment to greater than or equal to 35 atom % within 7 days of the receipt of the enrichment report. If, after this time period, the enrichment requirement is still not met, submit a report to the NRC within 7 days advising them of the plans to return the solution to greater than or equal to 35 atom % Boron-10 isotopic enrichment.

(2) A check shall be made to ensure that the sodium pentaborate solution meets the original design criteria by comparing the enrichment, concentration and volume to established criteria (Boron-10 equal to or greater than 82 pounds). If the sodium pentaborate solution does not meet the original criteria, be in the Shutdown condition within 24 hours.

3.4 EMERGENCY COOLING

Applicability: Applies to the operating status of the emergency cooling systems.

Objective: To assure operability of the emergency cooling systems.

Specifications:

A. Core Spray System

1. The core spray system shall be operable at all times with irradiated fuel in the reactor vessel, except as otherwise specified in this section.

2. The absorption chamber water volume shall be at least 82,000 $ft.^3$ in order for the core spray system to be considered operable.

3. If one core spray system loop or its core spray header delta P instrumentation becomes inoperable during the run mode, the reactor may remain in operation for a period not to exceed 7 days (See Note below) provided:

   a. The remaining loop has no inoperable components and is verified daily to be operable and,

   b. The average planar linear heat generation rate (APLHGR) of all the rods in any fuel assembly, as a function of average planar exposure, at any axial location shall not exceed 90% of the limits given in Specification 3.10.A. The action to bring the core to 90% of the APLHGR Limits must be completed within two hours after the system has been determined to be inoperable.

4. The reactor may remain in operation for a period not to exceed 15 days if one of the redundant active loop components in the core spray system becomes inoperable during the run mode provided:

   a. In the event of an inoperable core spray booster pump, the other core spray booster pump in the loop is verified daily to be operable.

   b. In the event of an inoperable core spray main pump, the other core spray main pump in the loop is verified daily to be operable and the APLHGR of all the rods in any fuel assembly, as a function of average planar exposure, at any axial location shall not exceed 90% of the limits given in Specification 3.10.A. The action to bring the core to 90% of the APLHGR Limits must be completed within two hours after the component has been determined to be inoperable.

automatic depressurization function) may be inoperable or bypassed during the system hydrostatic pressure test required by ASME Code Section XI, IS-500 at or near the end of each ten year inspection interval.

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. <u>Containment Spray System and Emergency Service Water System</u>

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.

1.1 Upon the accidental loss of secondary containment integrity, restored secondary containment integrity within 4 hours, or:

a. During Power Operation:

(1) Have the reactor mode switch in the shutdown mode position within the following 24 hours.

(2) Cease all work on the reactor or its connected systems in the reactor building which could result in inadvertent releases of radioactive materials.

(3) Cease all operations in, above or around the Spent Fuel Storage Pool that could cause release of radioactive materials.

b. During refueling:

(1) Cease fuel handling operations or activities which could reduce the shutdown margin (excluding reactor coolant temperature changes).

(2) Cease all work on the reactor or its connected systems in the reactor building which could result in inadvertent releases of radioactive materials.

(3) Cease all operations in, above or around the Spent Fuel Storage Pool that could cause release of radioactive materials.

2. Two separate and independent standby gas treatment system circuits shall be operable when secondary containment is required except as specified by Specification 3.5.B.3.

3. With one standby gas treatment system circuit inoperable:

a. During Power Operation:

(1) Verify the operability of the other standby gas treatment system circuit within 2 hours. If testing is required to demonstrate operability and significant painting, fire, or chemical release has taken place in the reactor building within the previous 12 hours, then demonstration by testing shall take place within 1 hour of the expiration of the 12 hour period, and

(2) Continue to verify the operability of the standby gas treatment system circuit once per 24 hours until the inoperable standby gas treatment circuit is returned to operable status.

(3) Restore the inoperable standby gas treatment circuit to operable status within 7 days or be subcritical with reactor coolant temperature less than 212°F within the next 36 hours.

b. During Refueling:

(1) Verify the operability of the other standby gas treatment system circuit within 2 hours. If testing is required to demonstrate operability and significant painting, fire, or chemical release has taken place in the reactor building within the previous 12 hours, then demonstration by testing shall take place within 1 hour of the expiration of the 12 hour period, and

(2) Continue to verify the operability of the redundant standby gas treatment system once per 7 days until the inoperable system is returned to operable status.

(3) Restore the inoperable standby gas treatment system to operable status within 30 days or cease all spent fuel handling, core alterations or operation that could reduce the shutdown margin (excluding reactor coolant temperature changes).

4. If Specifications 3.5.B.2 and 3.5.B.3 are not met, reactor shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours and the condition of Specification 3.5.B.1 shall be met.

Bases:

Specifications are placed on the operating status of the containment systems to assure their availability to control the release of any radioactive materials from irradiated fuel in the event of an accident condition. The primary containment system (1) provides a barrier against uncontrolled release of fission products to the environs in the event of a break in the reactor coolant systems.

Whenever the reactor coolant water temperature is above 212°F, failure of the reactor coolant system would cause rapid expulsion of the coolant from the reactor with an associated pressure rise in the primary containment. Primary containment is required, therefore, to contain the thermal energy of the expelled coolant and fission products which could be released from any fuel failures resulting from the accident. If the reactor coolant is not above 212°F, there would be no pressure rise in the containment. In addition, the coolant cannot be expelled at a rate which could cause fuel failure to occur before the core spray system restores cooling to the core. Primary containment is not needed while performing low power physics tests since procedures and the Rod Worth Minimizer would limit

3.8 ISOLATION CONDENSER

Applicability: Applies to operating status of the isolation condenser.

Objective: To assure heat removal capability under conditions of reactor vessel isolation from its normal heat sink.

Specification:

A. The two isolation condenser loops shall be operable during power operations and whenever the reactor coolant temperature is greater than 212°F except as specified in C, below or during reactor vessel pressure testing.

B. The shell side of each condenser shall contain a minimum water volume of 22, 730 gallons. If the minimum volume cannot be maintained or if a source of makeup water is not available to the condenser, the condenser shall be considered inoperable.

C. If one isolation condenser becomes inoperable during the run mode the reactor may remain in operation for a period not to exceed 7 days provided the motor operated isolation and condensate makeup valves in the operable isolation condenser are verified daily to be operable.

D. If Specification 3.8.A and 3.8.B are not met, or if an inoperable isolation condenser cannot be repaired within 7 days, the reactor shall be placed in the cold shutdown condition.

E. If an isolation condenser inlet (steam side) isolation valve (V-14-30, 31, 32 or 33) becomes or is made inoperable, in the open position during the run mode, the redundant inlet isolation valve shall be verified operable. If the inoperable valve is not returned to service within 4 hours declare the affected isolation condenser inoperable, isolate it and comply with Specification 3.8.C.

F. If an AC motor-operated isolation condenser outlet (condensate return) isolation valve (V-14-36 or 37) becomes or is made inoperable in the open position in the run mode, return the valve to service within 4 hours or declare the affected isolation condenser inoperable, isolate it and comply with Specification 3.8.C.

Basis: The purpose of the isolation condenser is to depressurize the reactor and to remove reactor decay heat in the event that the turbine generator and main condenser is unavailable as a heat sink.[1] Since the shell side of the isolation condensers operate at atmospheric pressure, they can accomplish their purpose when the reactor temperature is sufficiently above 212°F to provide for the heat transfer corresponding to reactor decay heat. The tube side of the isolation condensers form a closed loop with the reactor vessel and can operate without reducing the reactor coolant water inventory.

3.17 Control room Heating, Ventilating, and Air-Conditioning System

Applicability: Applies to the operability of the control room heating, ventilating, and air conditioning (HVAC) system.

Objective: To assure the capability of the control room HVAC system to minimize the amount of radioactivity from entering the control room in the event of an accident.

Specification:

A. The control room HVAC system shall be operable during all modes of plant operation.

B. With the control room HVAC system determined inoperable:

1. Verify once per 24 hours the partial recirculation mode of operation for the operable system, or place the operable system in the partial recirculation mode; and

2. Restore the inoperable system within 7 days, or prepare and submit a special report to the Commission in lieu of any other report required by Section 6.9, within the next 14 days, outlining the action taken, the cause of the inoperability and the plans/schedule for restoring the HVAC system to operable status.

C. With both control room HVAC systems determined inoperable.

1. During Power Operation: place the reactor in the cold shutdown condition with 30 hours

2. During Refueling:

(a) Cease irradiated fuel handling operations; and

(b) Cease all work on the reactor or its connected systems in the reactor building which could result in inadvertent releases of radioactive materials.

Basis: The operability of the control room HVAC system ensures that the control room will remain habitable for operations personnel during a postulated design basis accident. The control room envelope includes the control room panel area, the shift supervisor's office, toilet room, kitchen, and lower cable spreading room. Since Systems A and B do not have HEPA filters or charcoal absorbers, the supply fan and dampers for each system minimize the beta and gamma doses to the operators by providing positive pressurization and limiting the makeup and infiltration air into the control room envelope. For the supply of 100% outside air to the control room envelope, the dose increase to 29.1 rem beta and 3.14 rem gamma for the assumed 30 days, however, these values are within the allowable limits.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 167

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

By letter dated October 18, 1993, GPU Nuclear Corporation (GPUN/the licensee) submitted a request for changes to the Oyster Creek Nuclear Generating Station Technical Specification (TS). The request would revise TS 3.2C (Standby Liquid Control System), TS 3.4A (Core Spray System), TS 3.4B (Automatic Depressurization System), TS 3.4C (Containment Spray System and Emergency Service Water System), TS 3.5B (Secondary Containment), TS 3.8 (Isolation Condenser), and TS 3.17 (Control Room Heating, Ventilating, and Air-Conditioning System) to delete the current requirements to demonstrate by testing, that a redundant system/component is declared inoperable. These testing requirements would be replaced by requirements to verify that the redundant system/component is operable. These operability verifications would be accomplished by administrative checks of appropriate plant records (e.g., appropriate surveillance records and logs). Conforming changes would be made to Definition 1.1 (Operable-Operability).

2.0 EVALUATION

The requirement to demonstrate the operability, by testing, of a redundant system/component when a system/component is declared inoperable is a typical requirement that was included in the TS when Oyster Creek Nuclear Generating Station was granted its operating license. However, based on further operating experience, the NRC staff subsequently dropped such testing requirements. Testing of redundant systems/components is not required in the NRC's Standard Technical Specifications nor in recently issued TS. Deletion of such testing requirements was implemented by the NRC staff since the added operability assurance provided by such testing is not sufficient to justify the loss of safety function during the test, provided the periodic surveillance testing is current and that there are no known reasons to suggest that the redundant system/component is inoperable. The periodic surveillance tests and the proposed verifications that the redundant systems/components are

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operable are sufficient to demonstrate the operability of the redundant system/component. Therefore, the proposed changes to delete demonstration of operability by testing redundant system/components are acceptable.

3.0 STATE CONSULTATION

In a letter dated December 10, 1993, the State of New Jersey, Department of Environmental Protection and Energy - Division of Environmental Safety, Health and Analytical Programs, had the following comment in regard to GPUN's request to revise the Oyster Creek Technical Specifications for seven systems to delete the requirement for daily testing of redundant components when one train is inoperable.

> The determination of operability is an on-going process by plant operators. This concept was stressed by the NRC at the Operational Safety Team Inspection exit meeting at oyster Creek in October 1993. It should not be necessary to perform a special review of past surveillance tests and logs to verify operability. An effective surveillance testing program, accurate operating logs, timely preventive maintenance and other processes form the basis for operability decisions as these processes are performed. If GPU Nuclear and the NRC are satisfied with the performance of these existing programs at Oyster Creek, a special records review to verify operability should not be necessary.
>
> In reviewing NUREG 1433, Standard Technical Specifications for GE Plants, we could find no limiting condition that required a records review to determine operability. In addition, the definition of operability in this NUREG does not include the records review contained in the definition proposed by GPU Nuclear.

Staff's Response

Generic Letter 91-18 describes the relationship between surveillance requirements and the on-going process of assessing the operability of equipment. Older technical specifications included provisions to conduct tests on the alternate train of equipment when a train of equipment was determined to be inoperable. That position was later changed because such testing usually caused a loss of safety function. Instead, administrative procedures verified the operability of the alternate train. The same concept is reflected in the improved standard technical specifications (NUREG-1430 through NUREG-1434) in the administrative control for the Safety Function Determination Program (Section 5.8). That program provides the means to use plant status to determine whether inoperable equipment has caused a loss of safety function.

4.0 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 (58 FR 59749). 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.

5.0 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.

Principal Contributor: A. W. Dromerick

Date: December 21, 1993