1991

September 5, 19

Docket No. 50-219

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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

Dear Mr. Barton:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 77497)

The Commission has issued the enclosed Amendment No. 153 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated August 14, 1990, as revised June 18, 1991.

The amendment revises Technical Specification 3.4.A.3, 3.4.A.4, 3.4.D.2 and the associated bases of the Technical Specifications to incorporate the 10 CFR 50.46 loss-of-coolant accident analysis that is the basis for the MAPLHGR limits provided in the Technical Specification Section 3.10 "Core Limits." In your letter of June 18, 1991, you deleted the note to a more restrictive LCO for the Automatic Depressurization System (ADS) when one core spray loop is declared inoperable since the modification which eliminates the loss of the ADS was completed during the 13R refueling outage.

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. 153 to DPR-16
2. Safety Evaluation
3. Notice

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Mr. John J. Barton
GPU Nuclear Corporation

Oyster Creek Nuclear
Generating Station

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

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. 153
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 August 14, 1990, as revised June 18, 1991, 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) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 153, 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, 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 5, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 153

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
Page 3.4-1	Page 3.4-1
3.4-2	3.4-2
3.4-3	3.4-3
3.4-4	3.4-4
3.4-5	3.4-5
3.4-6	3.4-6
3.4-7	3.4-7
---	3.4-8

3.4 EMERGENCY COOLING

Applicability: Applies to the operating status of the emergency coolant 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 provided:

   a. The remaining loop has no inoperable components and is demonstrated 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 demonstrated 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 demonstrated 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.

If two of the redundant active loop components become inoperable, the limits of Specification 3.4.A.3 shall apply.

5. During the period when one diesel is inoperable, the core spray equipment connected to the operable diesel shall be operable.

6. If Specifications 3.4.A.3, 3.4.A.4, and 3.4.A.5 are not met, the reactor shall be placed in the cold shutdown condition. If the core spray system becomes inoperable, the reactor shall be placed in the cold shutdown condition and no work shall be performed on the reactor or its connected systems which could result in lowering the reactor water level to less than 4'8" above the top of the active fuel.

7. If necessary to accomplish maintenance or modifications to the core spray systems, their power supplies or water supplies, reduced system availability is permitted when the reactor is: (a) maintained in the cold shutdown condition or (b) in the refuel mode with the reactor coolant system maintained at less than 212^{O}F and vented, and (c) no work is performed on the reactor vessel and connected systems that could result in lowering the reactor water level to less than 4'8" above the top of the active fuel. Reduced Core Spray System Availability is minimally defined as follows:

    a. At least one core spray pump, and system components necessary to deliver rated core spray to the reactor vessel, must remain operable to the extent that the pump and any necessary valves can be started or operated from the control room or from local control stations.

    b. The fire protection system is operable, and

    c. These systems are demonstrated to be operable on a weekly basis.

8. If necessary to accomplish maintenance or modifications to the core spray systems, their power supplies or water supplies, reduced system availability is permitted when the reactor is in the refuel mode with the reactor coolant system maintained at less than 212^{O}F or in the startup mode for the purposes of low power physics testing. Reduced core spray system availability is defined as follows:

    a. At least one core spray pump in each loop, and system components necessary to deliver rated core spray to the reactor vessel, must remain operable to the extent that the pump and any necessary valves in each loop can be started or operated from the control room or from local control stations.

    b. The fire protection system is operable and,

    c. Each core spray pump and all components in 3.4.A.8a are demonstrated to be operable every 72 hours.

9. If Specifications 3.4.A.7 and 3.4.A.8 cannot be met, the requirements of Specification 3.4.A.6 will be met and work will be initiated to meet minimum operability requirements of 3.4.A.7 and 3.4.A.8.

10. The core spray system is not required to be operable when the following conditions are met:

   a. The reactor mode switch is locked in the "refuel" or "shutdown" position.

   b. (1) There is an operable flow path capable of taking suction from the condensate storage tank and transferring water to the reactor vessel, and

      (2) The fire protection system is operable.

   c. The reactor coolant system is maintained at less than 212°F and vented (except during reactor vessel pressure testing).

   d. At least one core spray pump, and system components necessary to deliver rated core spray flow to the reactor vessel, must remain operable to the extent that the pump and any necessary valves can be started or operated from the control room or from local control stations, and the torus is mechanically intact.

   e. (1) No work shall be performed on the reactor or its connected systems which could result in lowering the reactor water level to less than 4'8" above the top of the active fuel and the condensate storage tank level is greater than thirty (30) feet (360,000 gallons). At least two redundant systems including core spray pumps and system components must remain operable as defined in d. above

      OR

      (2) The reactor vessel head, fuel pool gate, and separator-dryer pool gates are removed and the water level is above elevation 117 feet.

      NOTE: When filling the reactor cavity from the condensate storage tank and draining the reactor cavity to the condensate storage tank, the 30 foot limit does not apply provided there is sufficient amount of water to complete the flooding operation.

B. <u>Automatic Depressurization System</u>

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 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 demonstrated 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 demonstrated 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 demonstrated 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.

7. If Specifications 3.4.C.3, 3.4.C.4, 3.4.C.5 or 3.4.C.6 are not met, the reactor shall be placed in cold shutdown condition. If the containment spray system or the emergency service water system becomes inoperable, the reactor shall be placed in the cold shutdown condition and no work shall be performed on the reactor or its connected systems which could result in lowering the reactor water level to less than 4'8" above the top of the active fuel.

8. The containment spray system may be made inoperable during the integrated primary containment leakage rate test required by Specification 4.5, provided that the reactor is maintained in the cold shutdown condition and that no work is performed on the reactor or its connected systems which could result in lowering the reactor level to less than 4'8" above the top of the active fuel.

D. Control Rod Drive Hydraulic System

1. The control rod drive (CRD) hydraulic system shall be operable when the reactor water temperature is above 212°F except as specified in 3.4.D.2 and 3.4.D.3 below.

2. If one CRD hydraulic pump becomes inoperable when the reactor water temperature is above 212°F, the reactor may remain in operation for a period not to exceed 7 days provided the second CRD hydraulic pump is operating and is checked at least once every 8 hours. If this condition cannot be met, the reactor water temperature shall be reduced to less than 212°F.

3. During reactor vessel pressure testing, at least one CRD pump shall be operable.

E. Core Spray and Containment Spray Pump Compartments Doors

The core spray and containment spray pump compartments doors shall be closed at all times except during passage in order to consider the core spray system and the containment spray system operable.

F. Fire Protection System

1. The fire protection system shall be operable at all times with fuel in the reactor vessel except as specified in Specification 3.4.F.2.

2. If the fire protection system becomes inoperable during the run mode, the reactor may remain in operation provided both core spray system loops are operable with no inoperable components.

Bases:

This specification assures operability of the emergency core cooling system to provide adequate core cooling. The Oyster Creek ECCS has two core spray loops; each containing a core spray sparger, two main pumps and two booster pumps. Specification 3.4.A.1 insures the availability of core cooling to meet the ECCS acceptance criteria in 10CFR50.46 utilizing the MAPLHGR limits provided in Section 3.10. These limits are from calculations(1) that include models and procedures which are specified in 10CFR50 Appendix K. A core spray flow of at least 3400 gpm (1 main and 1 booster pump) from 1 loop plus 2200 gpm (1 main pump) from the other loop at a vessel pressure of 110 psig is used in the calculation. Core spray loop 2 would be required to deliver 3640 gpm if loop 2 is relied upon as the two pump contributor and 2360 gpm if loop 2 is the single pump contributor, since loop 2 has flow losses through cracks in the core spray sparger.

Specification 3.4.A.3 allows continued operation with one core spray loop inoperable for a limited period of time. An evaluation of data presented in Reference 5 shows that flow from a single core spray sparger, main and booster pumps delivering 3400 gpm (3640 gpm for loop 2) at a vessel pressure of 110 psig, will meet 10 CFR 50.46 criteria with a 10% reduction in MAPLHGR Limits specified in Section 3.10. At 90% of the APLHGR, each core spray system is capable of supplying the required minimum bundle flow rate to ensure core cooling (References 6 and 7). Two hours is allowed for a reduction in the APLHGR limit which is consistent with two hours provided by Specification 3.10.A.3 to return an exceeded APLHGR to within the prescribed limit.

Under the operational constraints of Specification 3.4.A.3 the operable core spray loop meets all Appendix K requirements except for the case of a core spray line break inside the drywell in the operable loop. As a result, reactor operation is permitted for a period not to exceed seven days. The allowed time out of service for the redundant core spray loop is justified based on the low probability of the event, the direct operator indication of a Core Spray System pipe break, and emergency procedures which provide for additional cooling water through the fire system.

The probability of a pipe break between the reactor vessel and the core spray check valve in the operable core spray loop (approx. 28 feet of 6 inch pipe) compared to the total pipe in the reactor coolant pressure boundary is very small. The probability of a core spray line break in conjunction with the other core spray loop out of service, which in itself is a low probability, is so small that it does not constitute an unacceptable risk. In the extremely unlikely event that this LOCA scenario were to occur, the operators are provided with a specific visual and audible alarm alerting them of a "Core Spray System I (II) Pipe Break" (one for each core spray loop). These alarms are initiated by differential pressure detectors on each core spray loop. In such a case the core spray line break would occur above the top of the active fuel allowing the core to be re-flooded from the fire protection system through the intact core spray loop.

In addition, a small break LOCA in the operable core spray loop prior to a larger break will be detected by the drywell unidentified leakage system (drywell sump) even before it is detected by the core spray alarm system. This will provide the operators with additional time to respond.

Therefore, the out-of-service time for one of the two core spray loops, as evaluated as per the guidelines in Reference 8, has been conservatively selected to be 7 days.

Specification 3.4.A.4 allows continued operation with one component inoperable for a limited period of time. Each core spray loop contains redundant active components based upon Reference 1 or 5, as appropriate. Therefore, with the loss of one of these components, the system as a whole (both loops) can tolerate an additional single failure of one of its active components and still perform the intended function and meet 10CFR50.46 criteria. If a redundant active component fails, a fifteen day period is allowed for repairs, based on 1 out of 4 components being required. 3.4.A.4.b insures that the 1 out of 4 requirement is maintained.

Specification 3.4.A.5 ensures that if one diesel is out of service for repair, the core spray components fed by the other diesel must be operable. Since each diesel will provide power to components for both core spray loops, the required flow specified in the bases for Specification 3.1.A.1 will be met.

When the reactor is in the shutdown or refueling mode and the reactor coolant system is less than 212°F and vented and no work is being performed that could result in lowering the water level to less than 4'8" above the core, the likelihood of a leak or rupture leading to uncovering of the core is very low. The only source of energy that must be removed is decay heat and one day after shutdown this heat generation rate is conservatively calculated to be not more than 0.6% of rated power. Sufficient core spray flow to cool the core can be supplied by one core spray pump or one of the two fire protection system pumps under these conditions. When it is necessary to perform repairs on the core spray system components, power supplies or water sources, Specification 3.4.A.7 permits reduced cooling system capability to that which could provide sufficient core spray flow from two independent sources. Manual initiation of these systems is adequate since it can be easily accomplished within 15 minutes during which time the temperature rise in the reactor will not reach 2200°F.

In order to allow for certain primary system maintenance, which will include control rod drive repair, LPRM removal/installation, reactor leak test, etc., (all performed according to approved procedure), Specification 3.4.A.8 requires the availability of an additional core spray pump in an independent loop, while this maintenance is being performed the likelihood of the core being uncovered is still considered to be very low, however, the requirement of a second core spray pump capable of full rated flow and the 72 hour operability demonstration of both core spray pumps is specified.

Specification 3.4.A.10 allows the core spray system to be inoperable in the cold shutdown or refuel modes if the reactor cavity is flooded and the spent fuel pool gates are removed and a source of water supply to the reactor vessel is available. Water would then be available to keep the core flooded.

The relief valves of the automatic depressurization system enable the core spray system to provide protection against the small break in the event the feedwater system is not active.

The containment spray system is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. The flow from one pump in either loop is more than ample to provide the required heat removal capability(2). The emergency service water system provides cooling to the containment spray heat exchangers and, therefore, is required to provide the ultimate heat sink for the energy release in the event of a loss-of-coolant accident. The emergency service water pumping requirements are those which correspond to containment cooling heat exchanger performance implicit in the containment cooling description. Since the loss-of-coolant accident while in the cold shutdown condition would not require containment spray, the system may be deactivated to permit integrated leak rate testing of the primary containment while the reactor is in the cold shutdown condition.

The control rod drive hydraulic system can provide high pressure coolant injection capability. For break sizes up to 0.002 ft^2, a single control rod drive pump with a flow of 110 gpm is adequate for maintaining the water level nearly five feet above the core, thus alleviating the necessity for auto-relief actuation(3).

The core spray main pump compartments and containment spray pump compartments were provided with water-tight doors(4). Specification 3.4.E ensures that the doors are in place to perform their intended function.

Similarly, since a loss-of-coolant accident when primary containment integrity is not being maintained would not result in pressure build-up in the drywell or torus, the system may be made inoperable under these conditions. This prevents possible personnel injury associated with contact with chromated torus water.

References

1. NEDC-31462P, "Oyster Creek Nuclear Generating Station SAFER/CORECOOL/GESTR-LOCA Loss-of-Coolant Accident Analysis," August 1987.

2. Licensing Application, Amendment 32, Question 3

3. Licensing Application, Amendment 18, Question 1

4. Licensing Application, Amendment 18, Question 4

5. GPUN Topical Report 053, "Thermal Limits with One Core Spray Sparger" December 1988.

6. NEDE-30010A, "Performance Evaluation of the Oyster Creek Core Spray Sparger", January 1984.

7. Letter and enclosed Safety Evaluation, Walter A. Paulson (NRC) to P. B. Fiedler (GPUN), July 20, 1984.

8. APED-5736, "Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards", April 1969.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 153

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 licensee, GPU Nuclear Corporation (GPUN), by letters dated August 14, 1990 and June 18, 1991, proposed changes in Sections 3.4.A.3, 3.4.A.4, 3.4.D.2, and the associated Bases of the Technical Specification (TS) for the Oyster Creek Nuclear Generating Station (OCNGS). The proposed changes would revise the TS to incorporate the 10 CFR 50.46 loss-of-coolant accident (LOCA) analysis that is the basis for the Average Planar Linear Heat Generation Rate (APLHGR) limits provided in the TS Section 3.10, "Core Limits." The Limiting Conditions for Operation (LCOs) and Bases were changed as appropriate. The NRC staff has reviewed the safety implications of the proposed TS changes, and this evaluation presents a summary of the review.

2.0 EVALUATION

A specific listing and evaluation of the proposed changes are given below:

(1) TS 3.4.A.3: Part (b) is the only new addition in this section, which indicates that when one of the core spray loops becomes inoperable during the run mode, 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 percent of the limits given in Specification 3.10.A. The action to bring the core to 90 percent of the APLHGR limits must be completed within 2 hours after the system has been determined to be inoperable. This proposed change of lowering the APLHGR is conservative relative to the existing specification and, therefore, acceptable. Furthermore this proposed change is conservative in assuring that the peak clad temperature will remain below 2200°F under 10 CFR 50.46 Appendix K requirements when one core spray is out of service.

In the associated Bases, the licensee attempted to justify the use of the existing specification which allows reactor operation up to a period of 7 days when one of the core spray loops becomes inoperable during the run mode. The NRC staff acknowledges that the associated Bases are consistent with the current specification. However, we note that a break in the pipe between reactor vessel and the core spray check valve in one loop while the other loop is inoperable has the potential to lead to an unmitigated LOCA. Furthermore, the normal allowed time of reactor operation in such a degraded emergency core cooling system (ECCS) configuration is typically less than 7 days for other plants. Realizing the potential vulnerabilities that may exist in this particular segment of the plant, the licensee is examining the core damage frequency contribution of core spray line breaks separate from other LOCAs in the Individual Plant Examination (IPE) for the OCNGS. The results from this analysis can provide a technical basis for evaluating this issue via the licensee's IPE submittal. The NRC staff also recommends that due to the enhanced importance of the segment of the pipings between reactor vessel and core spray check valve in the core spray loops, this particular segment of the pipings be closely monitored for potential degradation.

(2) TS 3.4.A.4: This section has been revised to indicate that in the event of an inoperable core spray main pump, the other core spray main pump in the loop is demonstrated 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 percent of the limits given in Specification 3.10.A. The action to bring the core to 90 percent of the APLHGR limits must be completed within 2 hours after the component has been determined to be inoperable. The proposed changes are conservative and the value of 90 percent is based upon the use of approved models, meet applicable criteria and are acceptable.

(3) TS 3.4.D.2: This section has been revised to indicate that the symbol "<" be changed to "less than." This is merely an editorial change, and has no safety implications. The proposed change is acceptable.

As a result of our review, which is presented in the evaluation above, the NRC staff concludes that the proposed Technical Specification changes are acceptable.

3.0 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.

4.0 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 August 6, 1991 (56 FR 37372). Accordingly, based upon the environmental assessment, the NRC staff has determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

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: M. Razzaque

Date: September 5, 1991

UNITED STATES NUCLEAR REGULATORY COMMISSION

GPU NUCLEAR CORPORATION AND JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 153 to Facility 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 revises Technical Specification (TS) 3.4.A.3, 3.4.A.4, 3.4.D.2 and the associated Bases of the Technical Specifications to incorporate the 10 CFR 50.46 loss-of-coolant accident analysis that is the basis for the MAPLHGR limits provided in the TS Section 3.10 "Core Limits."

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 September 19, 1990 (55 FR 38620). 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 August 14, 1990, as supplemented June 18, 1991, (2) Amendment No. 153 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 NW., Washington, DC 20555 and at the local public document room located at the 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, DC 20555, Attention: Director, Division of Reactor Projects - I/II.

Dated at Rockville, Maryland this 5th day of September 1991.

FOR THE NUCLEAR REGULATORY COMMISSION

Alexander W. Dromerick

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