June 18, 1990

Docket No. 50-219

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

Dear Mr. Fitzpatrick:

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION - ISSUANCE OF AMENDMENT (TAC NO. 76363)

The Commission has issued the enclosed Amendment No. 140 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated March 19, 1990.

The amendment revises Technical Specification 3.3.F.2. Specifically, the change would include limitations on operation with an idle recirculation loop which is isolated. A revision to Section 3.3 and 3.10 bases would also be needed to reflect this change.

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. 140 to DPR-16
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	:LA:PDI-4	:PM:PDI-4	:PD:PDI-4	:OGC	:	:
NAME	:SNorris	:ADromerick:lm	:JStolz	:E Holler	:	:
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DATED: June 18, 1990

AMENDMENT NO. 140 TO FACILITY OPERATING LICENSE NO. DPR-16

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Mr. E. E. Fitzpatrick
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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 PROVISIONAL OPERATING LICENSE

Amendment No. 140
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 March 19, 1990 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Provisional Operating License No. DPR-16 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 140 , 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

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: June 18, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 140

PROVISIONAL 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 3.3-3	Page 3.3-3
Page 3.3-3a	Page 3.3-3a
Page 3.3-8	Page 3.3-8
Page 3.3-8a	Page 3.3-8a
Page 3.10-4	Page 3.10-4
Page 3.10-5	Page 3.10-5
Page 3.10-6	Page 3.10-6
	Page 3.10-6a

E. Reactor Coolant Quality

1. The reactor coolant quality during power operation with steaming rates to the turbine-condenser of less than 100,000 pounds per hour shall be limited to:

   conductivity 2 uS/cm [S=mhos at 25°C(77°F)]
   chloride ion 0.1 ppm

2. When the conductivity and chloride concentration limits given in 3.3.E.1 are exceeded, an orderly shutdown shall be initiated immediately, and the reactor coolant temperature shall be reduced to less than 212°F within 24 hours.

3. The reactor coolant quality during power operation with steaming rates to the turbine-condenser of greater than or equal to 100,000 pounds per hour shall be limited to:

   conductivity 10 uS/cm [S=mhos at 25°C(77°F)]
   chloride ion 0.5 ppm

4. When the maximum conductivity or chloride concentration limits given in 3.3.E.3 are exceeded, an orderly shutdown shall be initiated immediately, and the reactor coolant temperature shall be reduced to less than 212°F within 24 hours.

5. During power operation with steaming rates on the turbine-condenser of greater than or equal to 100,000 pounds per hour, the time limit above 1.0 uS/cm at 25°C (77°F) and 0.2 ppm chloride shall not exceed 72 hours for any single incident.

6. When the time limits for 3.3.E.5 are exceeded, an orderly shutdown shall be initiated within 4 hours.

F. Recirculation Loop Operability

1. During POWER OPERATION, all five recirculation loops shall be OPERATING except as specified in Specification 3.3.F.2.

2. POWER OPERATION with one idle recirculation loop or one fully isolated loop per F.2.c is permitted. When the idle loop is isolated the following conditions shall be met:

   a. The average planar linear heat generation rate (APLHGR) of all fuel rods in any fuel assembly, as a function of average planar exposure, at any axial location shall not exceed 98% of the limits given in the specifications for APLHGR in Section 3.10.A. The action to bring the core to 98% of the APLHGR limits shall be completed prior to isolating the recirculation loop.

   b. The associated recirculation pump motor generator set circuit breaker shall be opened and defeated from operation.

c. The suction valve, discharge valve and discharge bypass valve in the isolated loop shall be in the closed position and associated motor breakers shall be opened and defeated from operation.

d. The fully isolated loop as in 3.3.F.2.C above shall not be returned to service unless the reactor is in the COLD SHUTDOWN condition.

3. If Specifications 3.3.F.1 and 3.3.F.2 are not met, an orderly shutdown shall be initiated immediately until all operable control rods are fully inserted and the reactor is in either the REFUEL MODE or SHUTDOWN CONDITION within 12 hours.

4. With reactor coolant temperature greater then 212°F and irradiated fuel in the reactor vessel, at least one recirculation loop discharge valve and its associated suction valve shall be in the full open position.

5. If Specification 3.3.F.4 is not met, immediately open one recirculation loop discharge valve and its associated suction valve.

6. With reactor coolant temperature less than 212°F and irradiated fuel in the reactor vessel, at least one recirculation loop discharge valve and its associated suction valve shall be in the full open position unless the reactor vessel is flooded to a level above 185 inches TAF or unless the steam separator and dryer are removed.

pH, chloride, and other chemical parameters are made to determine the cause of the unusual conductivity and instigate proper corrective action. These can be done before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Several techniques are available to correct off-standard reactor water quality conditions including removal of impurities from reactor water by the cleanup system, reducing input of impurities causing off-standard conditions by reducing power and reducing the reactor coolant temperature to less than 212°F. The major benefit of reducing the reactor coolant temperature to less than 212°F is to reduce the temperature dependent corrosion rates and thereby provide time for the cleanup system to re-establlish proper water quality.

Specifications 3.3.F.1 and 3.3.F.2 require a minimum of four OPERATING recirculation loops during reactor POWER OPERATION. Core parameters have not been established for POWER OPERATION with less than four OPERATING loops. Therefore, Specification 3.3.F.3 requires reactor POWER OPERATION to be terminated and the reactor placed in the REFUEL MODE or SHUTDOWN CONDITION within 12 hours.

During four loop POWER OPERATION the idle loop, when it is not isolated, is required to have its discharge valve closed and its discharge bypass and suction valves open. This provides and limits reactor coolant backflow through an idle loop and thus minimizes the occurrence of a severe cold water addition transient during startup of an idle loop. In addition, with the discharge bypass and suction valves in an idle loop open the coolant inventory in the loop is available during LOCA blowdown.

The requirements of Specification 3.3.F.2 for partial loop operation in which the idle loop is isolated, preclude the inadvertent startup of a recirculation pump with a cold leg thus avoiding any reactivity addition transient or reactor vessel nozzle thermal stress concerns.

Specifications 3.3.F.4 and 3.3.F.6 assure that an adequate flow path exists from the annular space, between the pressure vessel wall and the core shroud, to the core region. This provides sufficient hydraulic communication between these areas, thus assuring that reactor water instrument readings are indicative of the level in the core region. For the bounding loss of feedwater transient(2), a single fully open recirculation loop transfers coolant from the annulus to the core region at approximately five times the boiloff rate with no forced circulation(3). With the reactor vessel flooded to a level above 185 inches TAF or when the steam separator and dryer are removed, the core region and all recirculation loops can therefore be isolated. When the steam separator and dryer are removed, safety limit 2.1.D ensures water level is maintained above the core shroud.

References: (1) FDSAR, Volume I, Section IV-2
(2) Letter to NRC dated May 19, 1979, "Transient of May 2, 1979"
(3) General Electric Co. Letter G-EN-9-55, "Revised Natural Circulation Flow Calculation", dated May 29, 1979
(4) Licensing Application Amendment 16, Design Requirements Section
(5) (Deleted)
(6) FDSAR, Volume I, Section IV-2.3.3 and Volume II, Appendix H
(7) FDSAR, Volume I, Table IV-2-1
(8) Licensing Application Amendment 34, Question 14
(9) Licensing Application Amendment 28, Item III-B-2
(10) Licensing Application Amendment 32, Question 15
(11) (Deleted)
(12) (Deleted)
(13) Licensing Application Amendment 16, Page 1
(14) GPUN TDR 725 Rev. 0: Testing and Evaluation of Irradiated Reactor Vessel Materials Surveillance Program Specimens

The maximum average planar LHGR limits of fuel types V and VB are shown in Figure 3.10-1 for five loop operation and in Figure 3.10-2 for four loop operation, and are the result of LOCA analyses performed by Exxon Nuclear Company utilizing an evaluation model developed by Exxon Nuclear Company in compliance with Appendix K to 10 CFR 50 (1). Operation is permitted with the four-loop limits of Figure 3.10-2 provided the fifth loop has its discharge valve closed and its bypass and suction valves open. Four loop operation is permitted with the idle loop isolated (suction, discharge and discharge bypass valves closed) with Exxon fuel assemblies since the Exxon assemblies are located only on the core periphery and operate at significantly lower MAPLHGR values than the rest of the core. The MAPLHGR multiplier in Figure 3.10-3 is further reduced for an isolated idle loop consistent with the multiplier for GE fuel. Additional requirements for isolated idle loop operation are given in Specification 3.3.F.2. In addition, the maximum average planar LHGR limits shown in Figures 3.10-1 and 3.10-2 for Type V and VB fuel were analyzed with 100% of the spray cooling coefficients specified in Appendix K to 10 CFR Part 50 for 7 x 7 fuel. These spray heat transfer coefficients were justified in the ENC Spray Cooling Heat Transfer Test Program (2).

The maximum average planar LHGR limits of fuel types P8x8R and GE8x8EB are shown in Figure 3.10-4 and Figure 3.10-5, for both 5-loop and 4-loop operation when the idle loop is not isolated, and are based on calculations employing the models described in Reference 4. Four loop operation is permitted with the idle loop isolated (suction, discharge and discharge bypass valves closed) provided that a MAPLHGR multiplier of 0.98 as shown in Reference 4, is applied to figures 3.10-4 and 3.10-5. Additional requirements for isolated idle loop operation are given in Specification 3.3.F.2. Power operation with LHGR's at or below those shown in Figures 3.10-4 and 3.10-5 assures that the peak cladding temperature following a postulated loss-of-coolant accident will not exceed the 2200°F limit.

The effect of axial power profile peak location for fuel types V and VB is evaluated for the worst break size by performing a series of fuel heat-up calculations. A set of multipliers is devised to reduce the allowable bottom skewed axial power peaks relative to center or above center peaked profiles. The major factors which lead to the lower MAPLHGR limits with bottom skewed axial power profiles are the change in canister quench time at the axial peak location and a deterioration in heat transfer during the extended downward flow period during blowdown. The MAPLHGR multiplier in Figure 3.10-3 shall only be applied to MAPLHGR determined by the evaluation model described in reference 1.

The possible effects of fuel pellet densification are:

1) creep collapse of the cladding due to axial gap formation;
2) increase in the LHGR because of pellet column shortening;
3) power spikes due to axial gap formation; and
4) changes in stored energy due to increased radial gap size.

Calculations show that clad collapse is conservatively predicted not to occur during the exposure lifetime of the fuel. Therefore, clad collapse is not considered in the analyses.

Since axial thermal expansion of the fuel pellets is greater than axial shrinkage due to densification, the analyses of peak clad temperatures do not consider any change in LHGR due to pellet column shortening. Although the formation of axial gaps might produce a local power spike at one location on any one rod in a fuel assembly the increase in local density would be on the order of only 2% at the axial midplane. Since small local variations in power distribution have a small effect on peak clad temperature, power spikes were not considered in the analysis of loss-of-coolant accidents (1).

Changes in gap size affect the peak clad temperatures by their effect on pellet clad thermal conductance and fuel pellet stored energy. Treatment of this effect combined with the effects of pellet cracking, relocation and subsequent gap closure are discussed in XN-174. Pellet-clad thermal conductance for Type V and VB fuel was calculated using the GAPEX model (XN-174).

The specification for local LHGR assures that the linear heat generation rate in any rod is less than the limiting linear heat generation rate even if fuel pellet densification is postulated. The power spike penalty for Type V and VB fuel is based on analyses presented in Facility Change Request No.6 and FDSAR Amendment No.76, respectively. The analysis assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with 95% confidence that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking.

The power spike penalty for GE fuel is described in Reference 3.

The loss of coolant accident (LOCA) analyses are performed using an initial core flow that is 70% of the rated value. The rationale for use of this value of flow is based on the possibility of achieving full power (100% rate power) at a reduced flow condition. The magnitude of the reduced flow is limited by the flow relationship for overpower scram. The low flow condition for the LOCA analysis ensures a conservative analysis because this initial condition is associated with a higher initial quality in the core relative to higher flow-lower quality conditions at full power. The high quality-low flow condition for the steady-state core operation results in rapid voiding of the core during the blowdown period of the LOCA. The rapid degradation of the coolant conditions due to voiding results in a decrease in the time to boiling transition and thus degradation of heat transfer with consequent higher peak cladding temperatures. Thus, analysis of the LOCA using 70% flow and 102% power provides a conservative basis for evaluation of the peak cladding temperature and the maximum average planar linear heat generation rate (MAPLHGR) for the reactor.

The APRM response is used to predict when the rod block occurs in the analysis of the rod withdrawal error transient. The transient rod position at the rod block and corresponding MCPR can be determined. The MCPR has been evaluated for different APRM responses which would result from changes in the APRM status as a consequence of bypassed APRM channel and/or failed/bypassed LPRM inputs. The steady state MCPR required to protect the minimum transient CPR of 1.07 for the worst case APRM status condition (APRM Status 1) is determined in the rod withdrawal error transient analysis. The steady state MCPR values for APRM status conditions 1, 2, and 3 will be evaluated each cycle.

The time interval of eight (8) hours to adjust the steady state of MCPR to account for a degradation in the APRM status is justified on the basis of instituting a control rod block which precludes the possibility of experiencing a rod withdrawal error transient since rod withdrawal is physically prevented. This time interval is adequate to allow the operator to either increase the MCPR to the appropriate value or to upgrade the status of the APRM system while in a condition which prevents the possibility of this transient occurring.

The steady-state MCPR limit was selected to provide margin to accommodate transients and uncertainties in monitoring the core operating state, manufacturing, and in the critical power correlation itself(3). This limit was derived by addition of the CPR for the most limiting abnormal operational transient caused by a single operator error or equipment malfunction to the fuel cladding integrity MCPR limit designated in Specification 2.1.

Transients analyzed each fuel cycle will be evaluated with respect to the steady-state MCPR limit specified in this specification.

The purpose of the K_f factor is to define operating limits at other than rated flow conditions. At less that 100% flow the required MCPR is the product of the operating limit MCPR and the K_f factor. Specifically, the K_f factor provides the required thermal margin to protect against a flow increase transient.

The K_f factor curves shown in Figure 3.10-6 were developed generically using the flow control line corresponding to rated thermal power at rated core flow and are applicable to all BWR/2, BWR/3 and BWR/4 reactors. For the manual flow control mode, the K_f factors were calculated such that at the maximum flow state (as limited by the pump scoop tube set point) and the corresponding core power (along the rated flow control line), the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPR's were calculated at different points along the rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR determines the value of K_f.

REFERENCES

(1) XN-75-55-(A), XN-75-55, Supplement 1-(A), XN-75-55. Supplement 2-(A), Revision 2, "Exxon Nuclear Company WREM-Based NJP-BWR ECCS Evaluation Model and Application to the Oyster Creek plant," April 1977.

(2) XN-75-36 (NP)-(A), XN-75-36 (NP) Supplement 1-(A), "Spray Cooling Heat Transfer phase Test Results, ENC - 8 x 8 BWR Fuel 60 and 63 Active Rods, Interim Report," October 1975.

(3) NEDE-24195; General Electric Reload Fuel Application for Oyster Creek.

(4) NEDE-31462P; "OYSTER CREEK NUCLEAR GENERATING STATION SAFER/CORECOOL/GESTR-LOCA LOSS-OF-COOLANT ACCIDENT ANALYSIS," August 1987.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 140

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 March 19, 1990 (Ref. 1), GPU Nuclear Corporation (the licensee) requested changes to the Oyster Creek Technical Specifications (TS). The TS change proposes to allow an idle recirculation loop to be isolated. Oyster Creek is currently authorized to continue power operation if one of the five recirculation loops is taken out of service provided the idle loop is not isolated. The allowed configuration is the loop discharge valve closed and the suction and discharge bypass valves open. The request would allow the discharge bypass valve and suction valve to be closed in addition to the discharge valve thus completely isolating the loop. The proposal will allow operation with only the discharge valve closed or with all valves closed.

GPU proposed the TS change as a result of a Technical Specification-required shutdown on February 6, 1990. The shutdown was required when the unidentified leak rate approached the Technical Specification limit of 5 gallons per minute. The leakage was due to a failed recirculation pump seal on recirculation loop A. If the pump could have been isolated, the leakage might have been reduced and a shutdown averted. The ability to isolate a recirculation pump would prevent unnecessary shutdowns in the future.

2.0 EVALUATION

There are two main differences in the LOCA analysis for four-recirculation-loop operation, as compared to the normal five-loop case:

(a) With fewer operating loops, each functioning loop will be carrying a higher percentage of the initial core recirculation flow. If a break in one loop occurs, then a faster core flow coastdown rate will result, which could yield an earlier boiling transition time.

(b) If the inactive loop is isolated by closing the inactive loop suction and discharge valves, there will be reduced coolant inventory, which may lead to earlier core uncovery during a LOCA.

The LOCA analyses for four loop operation were performed using the "Oyster Creek - SAFER/CORECOOL/GESTR LOCA" methodology (Ref. 2), which has been approved by the staff (Ref. 3). GE performed the analyses with one out of service loop's inventory subtracted from the water available for blowdown. The small break (0.01 ft^2) and the DBA were calculated with the reduced inventory. The effects of faster core flow coastdown rate were also considered in the analyses. The results from the analysis demonstrate that 10 CFR 50.46 criteria for peak cladding temperature and maximum cladding oxidation are met, provided that an appropriate MAPLHGR multiplier is applied to the initial reactor power conditions. These multipliers are given below:

Fuel Type	Exposure Range	MAPLHGR Multiplier
P8x8R	E $\leq$ 15.0 GWd/MTU	0.99
P8x8R	E $>$ 15.0 GWd/MTU	0.98
GE8x8EB	All Exposures	0.98

In addition to the GE fuel assemblies, there are 29 ANF fuel assemblies which were loaded in Cycle 10 or earlier that are now located on the core periphery. The ANF four loop analysis does not address an isolated recirculation loop. The MAPLHGR multiplier of 0.98 is also applied to ANF fuel. This is acceptable.

When a recirculation loop is fully isolated at power, the isolated portion between the suction and discharge valves will cool to near ambient temperature.

Before the pump in the fully isolated recirculation loop can be restarted, the loop temperature must be warmed to within 50°F of the bulk coolant temperature in order to avoid the injection of cold water into the reactor core to prevent a transient and to avoid thermal stresses to the reactor vessel nozzles and CRD housings. This requirement cannot be satisfied with the current system configuration. Therefore, a fully isolated loop will not be restarted once it is isolated unless the reactor is in the cold shutdown condition. The suction valve, discharge valve and discharge bypass valve in the fully isolated loop will be in the closed position and the associated motor breakers shall be opened and defeated to prevent cold water injection into the vessel. This is acceptable.

3.0 TECHNICAL SPECIFICATION CHANGES

Section 3.3.F.2

(a) MAPLHGR multiplier of 0.98 is applied to APLHGR sections in 3.10.A. This change is supported by the analytical results that demonstrate no violation to the fuel integrity acceptance criteria and the fuel performance criteria of 10 CFR 50.46. This is acceptable.

(b) The idle recirculation pump MG set circuit breaker will be tagged out. This will prevent inadvertent startup of idle loop. This is acceptable.

(c) The idle recirculation loop suction valve, discharge valve and discharge bypass valves will be kept closed to prevent pump seal leakage. Their motors breakers will be tagged out. This will also prevent inadvertent startup of idle loop. This is acceptable.

(d) The isolated loop will not be returned to service unless the reactor is in the cold shutdown condition. This will prevent thermal stress of the RCS. This is acceptable.

Bases Sections 3.3 and 3.10

The revisions properly reflect the affected changes and hence the changes are acceptable.

The proposed request to isolate the idle recirculation loop completely is acceptable as discussed in Section 2.0.

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 June 14, 1990 (55 FR 24170). Accordingly, based upon the environmental assessment, we have determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

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 (3) the issuance of the 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 dated March 19, 1990 from E. E. Fitzpatrick, Vice President and Director, Oyster Creek to USNRC.

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

3. Letter from A. Thadani (NRC) to H. Pfefferlen (GE), dated May 11, 1987.

Dated: June 18, 1990

Principal Contributor: George Thomas

UNITED STATES NUCLEAR REGULATORY COMMISSION

GPU NUCLEAR CORPORATION

AND JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

NOTICE OF ISSUANCE OF AMENDMENT TO

PROVISINAL OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No.140 to Provisional 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 3.3.F.2. Specifically, the change would include limitations on operation with an idle recirculation loop which is isolated. A revision to Section 3.3 and 3.10 bases would also be needed to reflect this change.

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 April 10, 1990 (55 FR 13341). 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 March 19, 1990, (2) Amendment No.140 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 N.W., Washington, D.C. and 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, D.C. 20555, Attention: Director, Division of Reactor Projects - I/II.

Dated at Rockville, Maryland this 18th day of June 1990.

FOR THE NUCLEAR REGULATORY COMMISSION

Alexander W. Dromerick

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