1991

February 20, 199

Docket No. 50-219

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Mr. J. J. Barton, Director
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Barton

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 76769)

The Commission has issued the enclosed Amendment No. 147 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated May 7, 1990, as supplemented September 14, 1990, and December 13, 1990.

The amendment modifies the Technical Specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to a Core Operating Limits Report.

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. 147 to DPR-16
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	:PDI-4:LA	:PDI-4:PM	:PDI-4:D	:OGC	:
NAME	:SNorris	:ADromerick/Bah	:JStolz	:	:
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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. 147
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 May 7, 1990, as supplemented September 14, 1990 and December 13, 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. 147, 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

Alexander W. Dromerick

for 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: February 20, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 147

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 i	Page i
Page 1.0-4	Page 1.0-4
Page 1.0-7	Page 1.0-7
--	Page 1.0-8
Page 3.10-1	Page 3.10-1
Page 3.10-2	Page 3.10-2
Page 3.10-3	Page 3.10-3
Page 3.10-4	Page 3.10-4
Page 3.10-5	--
Page 3.10-6	--
Page 3.10-6a	--
Page 3.10-7	--
Page 3.10-8	--
Page 3.10-9	--
Page 3.10-10	--
Page 3.10-11	--
Page 3.10-12	--
Page 6.12	Page 6.12
	Page 6.12a
	Page 6.12b

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1.19 INSTRUMENTATION SURVEILLANCE DEFINITIONS

A. Channel Check

A qualitative determination of acceptable operability by observation of channel behavior during operation. This determination shall include, where possible, comparison of the channel with other independent channels measuring the same variable.

B. Channel Test

Injection of a simulated signal into the channel to verify its proper response including, where applicable, alarm and/or trip initiating action.

C. Channel Calibration

Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm or trip.

D. Source Check

A SOURCE CHECK is the qualitative assessment of channel response when the channel sensor is exposed to a source of radioactivity.

1.20 FDSAR

Oyster Creek Unit No. 1 Facility Description and Safety Analysis Report as amended by revised pages and figure changes contained in Amendments 14, 31 and 45.*

1.21 CORE ALTERATION

A core alteration is the addition, removal, relocation or other manual movement of fuel or controls in the reactor core. Control rod movement with the control rod drive hydraulic system is not defined as a core alteration.

1.22 CRITICAL POWER RATIO

The critical power ratio is the ratio of that power in a fuel assembly which is calculated, by application of an NRC approved CPR correlation, to cause some point in that assembly to experience boiling transition divided by the actual assembly operating power.

1.23 STAGGERED TEST BASIS

A Staggered Test Basis shall consist of:

A. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.

*Per Errata dtd. 4-9-69

1.37 PURGE

PURGE OR PURGING is the controlled process of discharging air or gas from a confinement and replacing it with air or gas.

1.38 EXCLUSION AREA

EXCLUSION AREA is defined in 10 CFR part 100.3(2). As used in these technical specification, the Exclusion Area boundary is the perimeter line around the OCNGS beyond which the land is neither owned, leased, nor otherwise subject to control by GPU (ref. ODCM Figure 1-1). The area outside the Exclusion Area is termed OFFSITE.

1.39 REACTOR VESSEL PRESSURE TESTING

System pressure testing required by ASME Code Section XI, Article IWA-5000, including system leakage and hydrostatic test, with reactor vessel completely water solid, core not critical and section 3.2.A satisfied.

1.40 SUBSTANTIVE CHANGES

SUBSTANTIVE CHANGES are those which affect the activities associated with a document or the document's meaning or intent. Examples of non-substantive changes are: (1) correcting spelling, (2) adding (but not deleting) sign-off spaces, (3) blocking in notes, cautions, etc., (4) changes in corporate and personnel titles which do not reassign responsibilities and which are not referenced in the Appendix A Technical Specifications, and (5) changes in nomenclature or editorial changes which clearly do not change function, meaning or intent.

1.41 DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 microcuries per gram which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table E-7 or Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluences for the Purpose of Evaluating Compliance with 10 CFR Part 40 Appendix I".

1.42 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the heat generation rate per unit length of fuel rod for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle at that height.

1.43 CORE OPERATING LIMITS REPORT

The Oyster Creek CORE OPERATING LIMITS REPORT (COLR) is the document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.f. Plant operation within these operating limits is addressed in individual specifications.

1.44 LOCAL LINEAR HEAT GENERATION RATE

The LOCAL LINEAR HEAT GENERATION RATE (LLHGR) shall be applicable to a specific planar height and is equal to the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) at the specified height multiplied by the local peaking factor at that height.

3.10 CORE LIMITS

Applicability: Applies to core conditions required to meet the Final Acceptance Criteria for Emergency Core Cooling Performance.

Objective: To assure conformance to the peak clad temperature limitations during a postulated loss-of-coolant accident as specified in 10 CFR 50.46 (January 4, 1974) and to assure conformance to the operating limits for local linear heat generation rate and minimum critical power ratio.

Specification:

A. Average Planar LHGR

During power operation the maximum AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for each fuel type as a function of exposure shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT (COLR).

If at any time during power operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, action shall be initiated to bring the reactor to the cold shutdown condition within 36 hours. During this period surveillance and corresponding action shall continue until reactor operation is within the prescribed limits at which time power operation may be continued.

B. Local LHGR

During power operation, the Local LINEAR HEAT GENERATION RATE (LHGR) of any rod in any fuel assembly, at any axial location shall not exceed the maximum allowable LHGR limits specified in the COLR.

If at any time during operation it is determined by normal surveillance that the limiting value of LHGR is being exceeded, action shall be initiated to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, action shall be initiated to bring the reactor to the cold shutdown condition within 36 hours. During this period, surveillance and corresponding action shall continue until reactor operation is within the prescribed limits at which time power operation may be continued.

C. Minimum Critical Power Ratio (MCPR)

During steady state power operation the MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit as specified in the COLR.

When APRM status changes due to instrument failure (APRM or LPRM input failure), the MCPR requirement for the degraded condition shall be met within a time interval of eight (8) hours, provided that the control rod block is placed in operation during this interval.

For core flows other than rated, the nominal value for MCPR shall be increased by a factor of k_f, where k_f is specified in the COLR.

If at any time during power operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded for reasons other than instrument failure, action shall be initiated to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two [2] hours, action shall be initiated to bring the reactor to the cold shutdown condition within 36 hours. During this period, surveillance and corresponding action shall continue until reactor operation is within the prescribed limit at which time power operation may be continued.

Bases:

The Specification for average planar LHGR assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46. The analytical methods and assumptions used in evaluating the fuel design limits are presented in FSAR Chapter 4.

LOCA analyses are performed for each fuel design at selected exposure points to determined APLHGR limits that meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using GE calculational models which are consistent with the requirements of 10 CFR 50, Appendix K.

The PCT following a postulated LOCA is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. Since expected location variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on th verage linear heat generation e is sufficient to assure that calculated temperatures are below the limits specified in 10 CFR 50.46.

The maximum average planar LHGR limits for the various fuel types currently being used are specified in the COLR. The MAPLHGR limits for both five-loop and four-loop operation with the idle loop unisolated are shown. Four-loop operation with the idle loop isolated (suction, discharge and discharge bypass valves closed) requires that a MAPLHGR multiplier of 0.98 be applied to all fuel types. Additional requirements for isolated loop operation are given in Specification 3.3.F.2.

Fuel design evaluations are performed to demonstrate that the cladding 1% plastic strain and other fuel design limits are not exceeded during anticipated operational occurrences for operation with LHGRs up to the operating limit LHGR.

The analytical methods and assumptions used in evaluating the anticipated operational occurrences to establish the operating limit MCPR are presented in the FSAR, Chapters 4, 6 and 15 and in Technical Specification 6.9.1.f. To assure that the Safety Limit MCPR is not exceeded during any moderate frequency transient event, limiting transients have been analyzed to determine the largest reduction in Critical Power Ratio (CPR). The types of transients evaluated are pressurization, positive reactivity insertion and coolant temperature decrease. The operational MCPR limit is selected to provide margin to accommodate transients and uncertainties in monitoring the core operating state, manufacturing, and in the critical power correlation itself. This limit is derived by addition of the CPR for the most limiting transient to the safety limit MCPR designated in Specification 2.1.

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 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. For those cycles where the rod withdrawal error transient is not the most severe transient the MCPR Value for APRM status conditions 1, 2, and 3 will be the same and be equal to the limiting transient MCPR value.

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.

Transients analyzed each fuel cycle will be evaluated with respect to the operational MCPR limit specified in the COLR.

The purpose of the k_f factor is to define operating limits at other than rated flow conditions. At less than 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, as shown in the COLR, were developed generically using the flow control line corresponding to rated thermal power at rated core flow. 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.

(4) a summary of meteorological data collected during the year shall be included in the report submitted within 60 days after January 1 of each year. Alternatively, summary meteorological data may be retained by GPU Nuclear and made available to the NRC upon request.

e. Annual Radiological Environmental Report: A report of radiological environmental surveillance activities during each year shall be submitted before May 1 of the following year. Each report shall include the following information required in Specification 4.16 for radiological environmental surveillance:

(1) a summary description of the radiological environmental monitoring program,

(2) a map and a table of distances and directions (compass azimuth) of locations of sampling stations from the reactor,

(3) results of analyses of samples and of radiation measurements, (In the event some results are not available, the reasons shall be explained in the report. In the event the missing results are obtained, they shall be reported to the NRC as soon as is reasonable.)

(4) deviation(s) from the environmental sampling schedule in Table 4.16.1.

(5) identification of environmental samples analyzed when instrumentation was not capable of meeting detection capability in Table 4.16.2.

(6) a summary of the results of the land use survey.

(7) a summary of the results of licensee participation in an NRC approved inter-laboratory crosscheck program for environmental samples.

(8) results of dose evaluations to demonstrate compliance with 40 CFR Part 190.10a.

f. CORE OPERATING LIMITS REPORT (COLR)

1. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle for the following:

   a. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.10.A

   b. The K_f core flow adjustment factor for Specification 3.10.C.

   c. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.10.C

d. The LOCAL LINEAR HEAT GENERATION RATE (LLHGR) for Specification 3.10.B.

and shall be documented in the COLR.

2. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.

   a. GPU Nuclear (GPUN) Topical Report (TR) 020, Methods for the Analysis of Boiling Water Reactors Lattice Physics, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

   b. GPUN TR 021, Methods for the Analysis of Boiling Water Reactors Steady State Physics, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

   c. GPUN TR 033, Methods for the Generation of Core Kinetics Data for RETRAN-02, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

   d. GPUN TR 040, Steady-State and Quasi-Steady-State Methods Used in the Analysis of Accidents and Transients, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

   e. GPUN TR 045, BWR-2 Transient Analysis Model Using the Retran Code, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

   f. NEDE-31462P and NEDE-31462, Oyster Creek Nuclear Generating Station SAFER/CORECOOL/GESTR-LOCA Loss-of-Coolant Accident Analysis, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

   g. NEDE-24011, General Electric Standard Application for Reactor Fuel, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

   h. NEDE-24195, General Electric Reload Fuel Application for Oyster Creek, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

   i. 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

j. XN-7_36(NP)-(A); XN-75-36(NP), Suppl_ent 1-(A), "Spray Cooling Heat Transfer Phase Test Results, ENC - 8x8 BWR Fuel 60 and 63 Active Rods, Interim Report," October 1975

3. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

4. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

Basis: 6.9.1.e

An annual report of radiological environmental surveillance activities includes factual data summarizing results of activities required by the surveillance program. In order to aid interpretation of the data, GPUN may choose to submit analysis of trends and comparative non regional radiological environmental data. In addition, the licensee may choose to discuss previous radiological environmental data as well as the observed radiological environmental impacts of station operation (if any) on the environment.

6.9.2 REPORTABLE EVENTS

The submittal of Licensee Event Reports shall be accomplished in accordance with the requirements set forth in 10 CFR 50.73.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 147

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 May 7, 1990 (Ref. 1) as supplemented by letters dated September 14, 1990 (Ref. 2), and December 13, 1990 (Ref. 3), GPU Nuclear Corporation (the licensee) proposed changes to the Technical Specifications (TS) for the Oyster Creek Nuclear Generating Station (OCNGS). The information provided in the September 14, 1990 and December 13, 1980 letters are in accordance with the guidance contained in Generic Letter 88-16 and have no effect on the no significant hazards consideration conclusions. The proposed changes would modify specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to a Core Operating Limits Report (COLR). The proposed changes also include the addition of the COLR to the Definitions section and to the reporting requirements of the Administrative Controls section of TS. Guidance on the proposed changes was developed by NRC on the basis of the review of a lead-plant proposal submitted on the Oconee Plant docket by Duke Power Company. This guidance was provided to all power reactor licensees and applicants by Generic Letter 88-16, dated October 4, 1988 (Ref. 4).

2.0 EVALUATION

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

(1) The Definitions section of the TS would be modified to include a definition of the COLR that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with NRC-approved methodologies that maintain the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

(2) The following specifications were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

(a) Specification 3.10.A

The Average Planar Linear Heat Generation Rate (APLHGR) limits for this specification are specified in the COLR (Figures 1 through 7).

(b) Specification 3.10.B

The local Linear Heat Generation Rate (LHGR) limits for this specification are specified in the COLR (Figure 10).

(c) Specification 3.10.C

The Minimum Critical Power Ratio (MCPR) limits and the MCPR core flow factor (K_f) for this specification are specified in the COLR (Figure 8 and Figure 9, respectively).

These changes to the specifications also required changes to the bases. Based on our review, we conclude that the changes to the bases are acceptable.

(3) Specification 6.9.1.f was added to the reporting requirements of the Administrative Controls section of the TS. This specification requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, this specification requires that the values of these limits be established using an NRC-approved methodology and be consistent with all applicable limits of the safety analysis. The approved methodology is the following:

(a) GPU Nuclear (GPUN) Topical Report (TR) 020, Methods for the Analysis of Boiling Water Reactors Lattice Physics (The approved revision at the time reload analyses analyses are performed shall be identified in the COLR.)

(b) GPUN TR 021, Methods for the Analysis of Boiling Water Reactors Steady State Physics (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

(c) GPUN TR 033, Methods for the Generation of Core Kinetics Data for RETRAN-02 (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

(d) GPUN TR 040, Steady-State and Quasi-Steady-State Methods Used in the Analysis of Accidents and Transients (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

(e) GPUN TR 045, BWR-2 Transient Analysis Model Using the Retran Code (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

(f) NEDE-31462P and NEDE-31462, Oyster Creek Nuclear Generating Station SAFER/CORECOOL/GESTR-LOCA Loss-of-Coolant Accident Analysis (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

(g) NEDE-24011, General Electric Standard Application for Reactor Fuel (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

(h) NEDE-24195, General Electric Reload Fuel Application for Oyster Creek (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

(i) 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

(j) XN-75-36(NP)-(A); XN-75-36(NP), Supplement 1-(A), "Spray Cooling Heat Transfer Phase Test Results, ENC - 8x8 BWR Fuel 60 and 63 Active Rods, Interim Report," October 1975

Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC, prior to operation with the new parameter limits.

On the basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in Generic Letter 88-16 on Modifying cycle-specific parameter limits in TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using NRC approved methodologies, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

As part of the implementation of Generic Letter 88-16, the staff has also reviewed a sample COLR that was provided by the licensee. On the basis of this review, the staff concludes that the format and content of the sample COLR are acceptable.

We have reviewed the request by the GPU Nuclear Corporation to modify the Technical Specifications of the Oyster Creek Nuclear Generating Station that would remove the specific values of some cycle-dependent parameters from the specifications and place the values in a COLR that would be referenced by the specifications. Based on this review, we conclude that these Technical Specification modifications are acceptable because they are in accordance with Generic Letter 88-16.

3.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The amendment also relates to changes in recordkeeping, reporting, or administrative procedures or requirements. We have 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 staff has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 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.

5.0 REFERENCES

1. Letter from E. E. Fitzpatrick (GPUN) to NRC, dated May 7, 1990.

2. Letter from E. E. Fitzpatrick (GPUN) to NRC, dated September 14, 1990.

3. Letter from M. W. Laggart (GPUN) to NRC, dated December 13, 1990.

4. Generic Letter 88-16, "Removal of Cycle-Specificat Parameter Limits from Technical Specifications," dated October 4, 1988.

Principal Contributor: T. Huang

Dated: February 20, 1991