October 31, 1988

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: ISSUANCE OF AMENDMENT (TAC NO. 67743)

The Commission has issued the enclosed Amendment No. 129 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated March 30, 1988, as supplemented April 12, 1988 and September 22, 1988.

The amendment modifies Section 3.10 of the Technical Specifications to accommodate the Cycle 12 Core Reload. Specifically, the Minimum Critical Power Ratio (MCPR) and the maximum average planar linear heat generator rated (MAPLHGR) limit is changed. It also permits the use of GE8x8EB fuel.

A copy of the related Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

orginal signed by John F. Stolz for

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

Enclosures:
1. Amendment No. 129 to DPR-16
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. E. E. Fitzpatrick
Oyster Creek Nuclear Generating Station

Oyster Creek Nuclear
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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.129
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 30, 1988, as supplemented April 12, 1988 and September 22, 1988 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. 129, 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.

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: October 31, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 129

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 3.10-1	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 3.10-5
Page 3.10-6	Page 3.10-6
Page 3.10-10	Page 3.10-10
Page 3.10-11	Page 3.10-11

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 14.5 KW/ft (for V and VB fuel and 13.4 KW/ft (for P8x8R and GE8x8EB fuel) operating limits for local linear heat generation rate.

Specification: A. Average Planar LHGR

During power operation, the average linear heat generation rate (LHGR) of all the rods in any fuel assembly, as a function of average planar exposure, at any axial location shall not exceed:

A.1 Fuel Types V and VB

The product of the maximum average planar LHGR (MAPLHGR) limit shown in Figures 3.10-1 (for 5-loop operation) and 3.10-2 (for 4-loop operation) and the axial MAPLHGR multiplier in Figure 3.10-3.

A.2 Fuel Types P8x8R and GE8x8EB

The maximum average planar LHGR (MAPLHGR) limit shown in Figure 3.10-4 and 3.10-5 for both 5-loop and 4-loop operation.

A.3 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 linear heat generation rate (LHGR) of any rod in any fuel assembly, at any axial location shall not exceed the maximum allowable LHGR:

B.1 Fuel Types V and VB

As calculated by the following equation;

$$LHGR \leq LHGR_d \left[1 - \frac{\Delta P}{P} \max \left(\frac{L}{LT} \right) \right]$$

Where: $LHGR_d$ = Limiting LHGR (=14.5)

$\frac{\Delta P}{P}$ = Maximum Power Spiking Penalty (=0.033 and 0.039 for fuel Types V and VB respectively)

LT = Total Core Length - 144 inches
L = Axial position above bottom of core

B.2 Fuel Type P8x8R and GE8x8EB

LHGR $\leq$ 13.4 KW/ft.

B.3 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, MCPR shall be greater than or equal to the following:

	APRM STATUS	MCPR Limit
1.	If any two (2) LPRM assemblies which are input to the APRM system and are separated in distance by less than three (3) times the control rod pitch contain a combination of three (3) out of four (4) detectors located in either the A and B or C and D levels which are failed or bypassed i.e., APRM channel or LPRM input bypassed or inoperable.	1.51

	APRM STATUS	MCPR Limit
2.	If any LPRM input to the APRM system at the B, C, or D level is failed or bypassed or any APRM channel is inoperable (or bypassed).	1.51
3.	All B, C, and D LPRM inputs to the APRM system are operating and no APRM channels are inoperable or by passed.	1.51

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 as shown in Figure 3.10-6.

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 (January 4, 1974) considering the postulated effects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on 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 the average linear heat generation rate is sufficient to assure that calculated temperatures are below the limits specified in 10 CFR 50.46 (January 4, 1974).

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. 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, and are based on calculations employing the models described in Reference 4. 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 valves 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 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 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

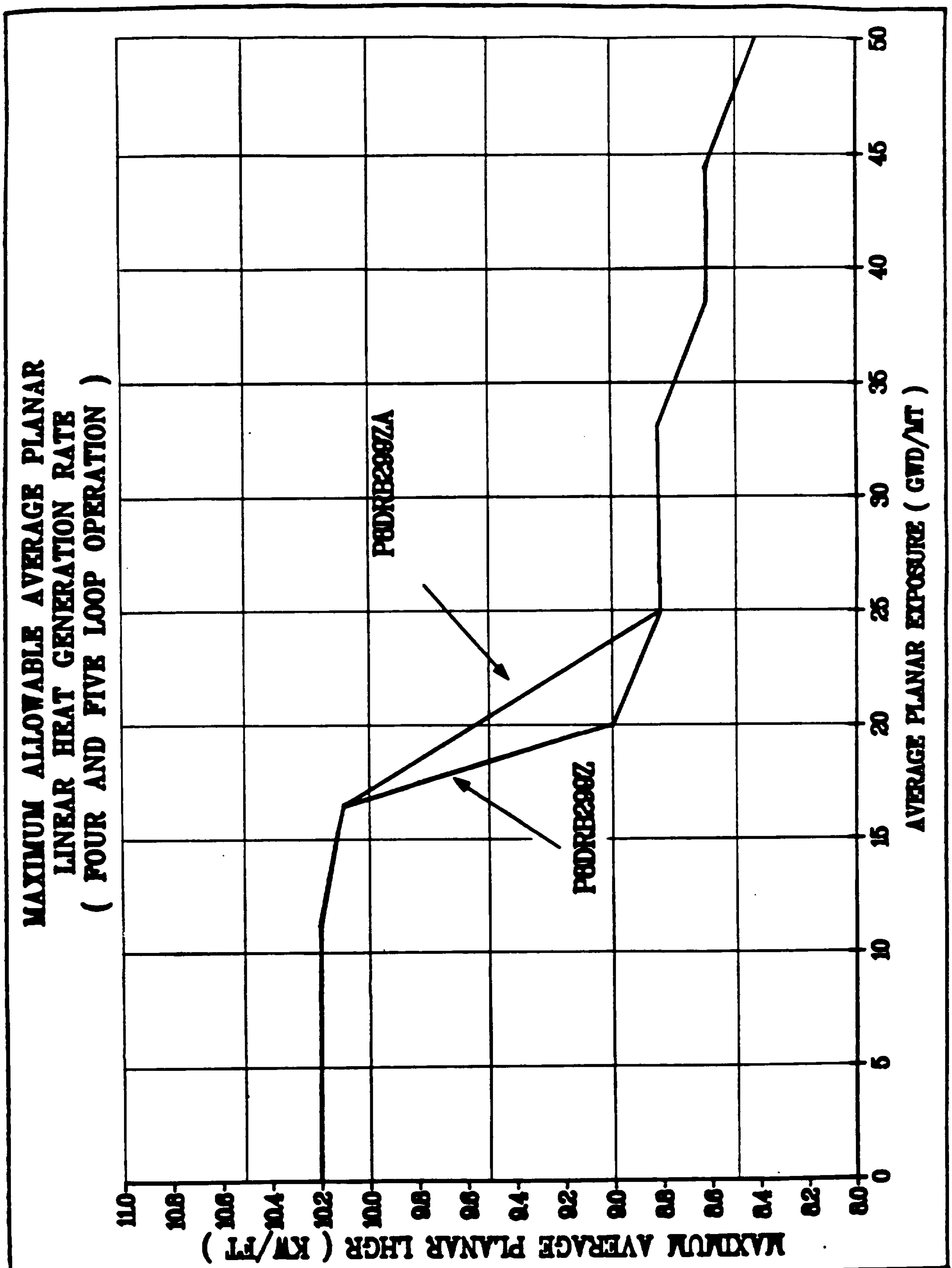
(1) XN-75-55-(A), XN-75-55, Supplement 1-(A), XN-75-55. Supplement 2-(A), Revision 2, "Exxon Nuclear Company WREM-Based NJP-BWR ECCS Evaluation Model and Application to the Oyster Creek plant," April 1977.

(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.

FIGURE 3.10-4


MAXIMUM ALLOWABLE AVERAGE PLANAR
LINEAR HEAT GENERATION RATE
(FOUR AND FIVE LOOP OPERATION)
P8DRB299ZA
P8DRB299Z
0
5
10
15
20
25
30
35
40
45
50
AVERAGE PLANAR EXPOSURE (GWD/MT)
11.0
10.8
10.6
10.4
10.2
10.0
9.8
9.6
9.4
9.2
9.0
8.8
8.6
8.4
8.2
8.0
MAXIMUM AVERAGE PLANAR LHGR (KW/FT)

MAXIMUN ALLOWABLE AVERAGE PLANAR LINEAR HEAT GENERATION RATE (FOUR AND FIVE LOOP OPERATION)

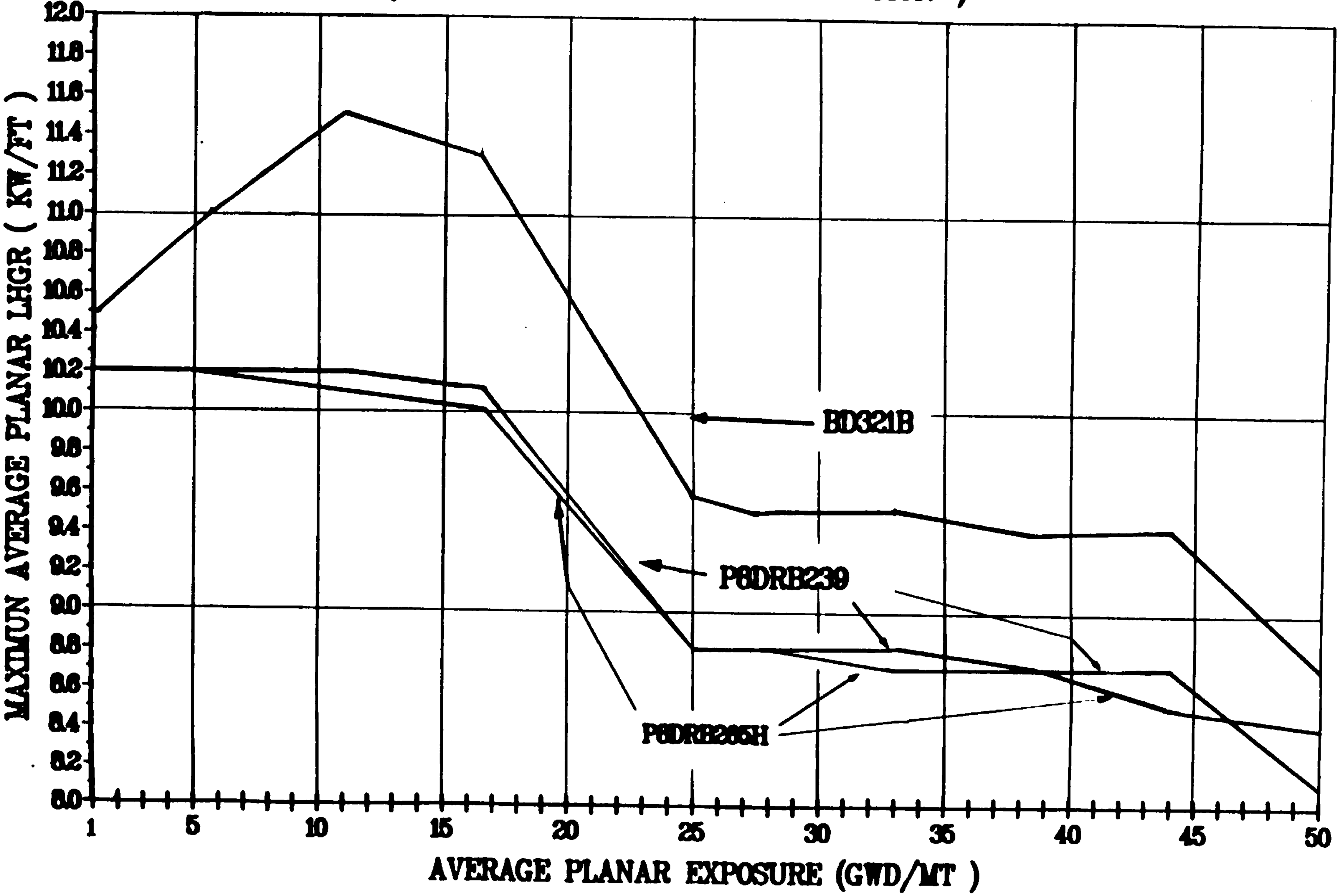


FIGURE 3.10-5

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 129

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

INTRODUCTION

By letters dated March 30, 1988, April 12, 1988 and September 22, 1988 (Ref. 1), GPU Nuclear Corporation (GPUN) made application to amend to Technical Specifications of the Provisional Operating License DPR-16 for the Oyster Creek Plant in order to operate for Cycle 12. In support of this application, the licensee also provided a reload analysis submittal (Ref. 2) and supporting analyses (Ref. 10). The reload application involves 3 fuel-design related issues: (1) the replacement of 172 spent fuel assemblies with 20 General Electric P8X8R and 152 GE8x8EB (extended burnup) fuel assemblies, (2) the analysis of safety considerations involved in the determination of Cycle 12 operating limits, and (3) the incorporation of new and extended maximum average planar linear heat generation rate (MAPLHR) limits. The NRC staff has reviewed these submittals as follows.

2.0 FUEL DESIGN

The Oyster Creek Cycle 12 core (see Table 1 for the fuel inventory) will retain 28 Exxon Type VB assemblies and 360 General Electric (GE) P8x8R assemblies from the previous cycle and add 20 unirradiated GEP8x8R fuels and 152 new GE8x8EB fuels, which are 3.21 percent average U235 enriched fuel assemblies. The GE8x8EB fuel type was approved in the Safety Evaluation (Ref. 13) for Amendment 10 to GESTAR II (Ref. 14) and has been used in many existing GE plants. The specific descriptions of this fuel were also included in Amendment 18 to GESTAR II (Ref. 16) which was previously approved by NRC (Ref. 15). LOCA analyses have been done for the retained and reload GE fuel using the SAFER/GESTAR-LOCA methods approved by the NRC (see Section 6.0). Since the MAPLHGR values for the fuel assemblies have been calculated with approved methodology (GESTAR II, Reference 14, Section 2 of Volume 1) they are acceptable.

The proposed average linear heat generation rate (LHGR) limit for the GE8x8EB fuel is 13.4 Kw/ft, which is less than the limit of 14.4 Kw/ft approved by the NRC for the GE extended burnup fuel (Ref. 13). The LHGR is, therefore, acceptable for the GE8x8EB fuel in the Oyster Creek Cycle 12 core.

3.0 NUCLEAR DESIGN

The nuclear design for Oyster Creek Cycle 12 has been performed by GPUN with the approved methodologies, which include fuel lattice methods (Refs. 3 and 4) and 3-dimensional core steady state methods (Refs. 5 and 6). The fuel lattice methods are used to calculate fuel bundle nuclear parameters such as reactivities, relative rod powers and 2 or 4 group cross sections. The 3-dimensional reactor code calculates power and exposure distributions, core thermal-hydraulic characteristics, and cold shutdown margin. The results of the reload analyses are given in Reference 2. The results are within the range of those usually encountered for BWR reloads. In particular, the shutdown margin is .0161 delta K at the exposure of minimum shutdown margin thus, fully meeting the required 0.01 delta K. The Standby Liquid Control System also meets shutdown margin requirements with a shutdown margin of 0.034 delta K. Since the Oyster Creek Cycle 12 nuclear design parameters have been obtained with previously approved methods and fall within expected ranges, the nuclear design is acceptable.

4.0 THERMAL AND HYDRAULIC DESIGN

The objective of the review is to confirm that the thermal-hydraulic design of the core has been accomplished using acceptable methods, and that it provides an acceptable margin of safety from conditions which could lead to fuel damage during transient conditions. The review includes two areas: safety limit minimum critical power (MCPR) and operating MCPR limits.

The licensee has submitted the analysis report for Cycle 12 operation at rated flow conditions (Ref. 2). Discussion of the review concerning the thermal-hydraulic design for the Cycle 12 operation follows:

4.1 Safety Limit MCPR

A safety limit MCPR has been imposed to assure that 99.9 percent of the fuel rods in the core are not expected to experience boiling transition during operational transients. As stated in Reference 20, a safety limit MCPR of 1.04 was approved to be applied to the second successive reload core of P8x8R, BP8x8R, GE8x8R or GE8x8EB fuel designs with an initial bundle R factor greater than or equal to 1.04. To provide more safety margin, the safety limit of 1.07 is used by GPUN for Oyster Creek Cycle 12 reload analysis.

4.2 Operating Limit MCPR

The most limiting events have been analyzed by the licensee to determine which event could potentially induce the largest reduction in the initial critical power ratio (RCPR). The RCPR values given in Table 5.1 of Reference 2 are plant specific values calculated by the methods including NODE-B (Ref. 8) and RETRAN (Ref. 9) methods, which were previously approved (Refs. 17 through 19) by NRC for the Oyster Creek reload applications. The turbine trip without bypass event was identified as the worst case with the largest RCPR of 0.37.

The proposed operating limit MCPR of 1.51, a more restrictive operating limit as compared to 1.45 for Cycle 11, was determined by adding the safety limit MCPR of 1.07 to a maximum RCPR of 0.37 with inclusion of statistical uncertainty factor of 1.049. Since the approved methods were used to determine the operating limit MCPR to avoid violation of the safety limit MCPR in the event of any anticipated transients, we conclude that the proposed operating limit MCPR of 1.51 is acceptable for incorporation into the Technical Specifications.

5.0 TRANSIENT ANALYSES

The transient analysis methodology used for the Oyster Creek Cycle 12 core described in References 8 and 9 were previously approved (Refs. 17, 18 and 19) by NRC. The CPM and NODE-B codes and methods described in Reference 8 were used to analyze the non-pressurization events including the loss of feedwater event. The RETRAN code and methods described in Reference 9 were used to analyze the pressurization transients including the turbine trip without bypass, loss of feedwater heater, feedwater controller failure and main steam isolation valve closure without scram. The limiting MCPR event for the Oyster Creek Cycle 12 core is the turbine trip without bypass and is discussed in Section 4.0. Compliance with overpressurization criteria was demonstrated by analysis of main steam isolation valve closure without credit for the first safety grade scram signals. Maximum vessel pressure was 1305 psia, which is well under 110 percent (1390 psia) of design pressure. Since bank position withdrawal sequence and rod pattern are used for Oyster Creek, a cycle specific control rod drop accident analysis is not required. The basis for this position and NRC approval is presented in Amendment 9 to Reference 14. We find that the approved methodologies and analytical results for both the pressurization and non-pressurization events were used to show that the analytical results fall within the safety limit to achieve the fuel and pressure boundary integrity during transients, therefore, we conclude the transient analyses are acceptable.

6.0 LOCA ANALYSES

The LOCA analyses for the Oyster Creek Cycle 12 core were performed using the SAFER/CORECOOL/GESTR LOCA methodology (Ref. 11), which has been approved by the staff (Ref. 12) and used and approved in several jet pump BWR reload applications (e.g., Duane Arnold Cycle 9, Quad Cities 1 Cycle 10). In Reference 12, the staff has specified conditions for demonstrating applicability of the SAFER/CORECOOL/GESTR LOCA methodology. These conditions are:

1. Calculation of a sufficient number of plant specific Peak Clad Temperature (PCT) points based on both nominal input values and Appendix K values to verify the shape of the PCT curves versus break size.

2. Confirmation that plant specific operating parameters have been bounded by the models and input used in generic calculations.

The licensee has provided the results of those analyses (Refs. 1 and 10) which are required to meet these conditions. Specifically, the analyses include break sizes from 0.05 ft^2 to the design basis accident (DBA) recirculation discharge line break (4.66 ft^2). Six different break sizes were analyzed in conjunction with ECCS failure combinations. A total of 15 cases were evaluated to establish the trend of PCT curves (for both the nominal and Appendix K conditions) versus break size.

The input parameters for both the nominal and Appendix K cases are within those used in the approved generic analyses. The results show that the DBA recirculation line discharge break with automatic depressurization system (ADS) valve failure is the limiting case. The calculated PCTs are 1831°F and 1714°F for low and high exposures, respectively, when nominal input values are used. The corresponding PCTs for this break size with Appendix K input values were calculated to be 2196°F and 2027°F for the low and high exposures, respectively. Because the approved methods were used in the analysis, the input parameters and the cases analyzed to establish the trend of PCT versus break size meet the staff requirements given above, and the Appendix K SAFER/CORECOOL results (which are 2196°F and 2027°F) bound the generic upper limit PCTs (which are 2088°F and 1999°F, and approved by NRC in Reference 11), we conclude that these analyses are acceptable.

7.0 TECHNICAL SPECIFICATION CHANGES

Various changes to Technical Specification (TS) 3.10 have been proposed in order for GPUN to operate the Oyster Creek Plant, Cycle 12 core. These changes, indicated in the proposed TS from pages 3.10-1 through 3.10.6, pages 3.10-10 and 3.10-11 (Ref. 1), range from miscellaneous changes (i.e., adding new references, etc.) to new and extended MAPLHGR limits, and operating limit MCPR for Cycle 12 fuels.

The Technical Specifications changes are:

1. Section 3.10.A: Add new limits for GE8x8EB fuel and revise limits for P8x8R fuel design. The new and revised MAPLHGR will be applied to both four loop (with the inactive loop suction and discharged valves unisolated) and five loop operation.

2. Section 3.10.B: Add reference for new fuel design (GE8x8EB to include LHGR limit of equal to or less than 13.4 kw/ft.

3. Section 3.10.C: Change operating limits MCPR from 1.45 to 1.51 for each of the three APRM status levels.

We find that all of these TS changes reflect the characteristics of fuels in Cycle 12 and are supported by the analytical results that demonstrate no violation to the fuel integrity acceptance criteria and the fuel performance acceptance criteria of 10 CFR 50.46, and therefore, we conclude the TS changes acceptable.

8.0 CONCLUSIONS

We have reviewed the reports submitted for the Cycle 12 reload of Oyster Creek with GE fuel, and the GE methodology and analysis for LOCAs, and the GPUN methodology and analysis for transients. Based on this review, we concluded that appropriate material was submitted and that the fuel design, nuclear design, thermal hydraulic design and transient and accident analyses are acceptable. The TS changes submitted for this reload suitably reflect the use of acceptable methodologies. The operating limits associated with those changes and reload parameters are acceptable.

9.0 ENVIRONMENTAL CONSIDERATION

This 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 staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

10.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security nor to the health and safety of the public.

11.0 REFERENCES

1. Letters from R. F. Wilson (GPU) to Document Control Center of NRC, Oyster Creek Nuclear Generating Station (Docket No. 50-219) - Technical Specification change Request (TSCR) No. 166, dated March 30, 1988, April 12, 1988 and September 22, 1988.

2. GPU TR-049 (Rev. 0), Reload Information and Safety Analysis Report for Oyster Creek Cycle 12 Reload, March 1988.

3. GPU TR-020-A (Rev. 0) Methods for the Analysis of Boiling Water Reactors Lattice Physics, January 1988.

4. Letter from J. Donohew (NRC) to P. Fiedler (GPU), Response Topical Report TR 020 (TAC 60339), November 14, 1986.

5. GPU-TR 021-A (Rev. 0), Methods for the Analysis of Boiling Water Reactors Steady State Physics, January 1988.

6. Letter from A. Dromerick (NRC) to P. Fiedler (GPU), dated September 27, 1987.

7. GPU-TR 033 (Rev. 0), Methods for Generation of Core Kinetics Data for RETRAN-02, February 1987.

8. GPU-TR 040 (Rev. 0), Steady-State and Quasi-Steady-State Methods Used in the Analysis of Accidents and Transients, February 1987.

9. GPU-TR 045 (Rev. 0), BWR-2 Transient Analysis Model Using the RETRAN Code, September 1987.

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

11. NEDE-30996-P, Volume II, "SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet and Non-Jet Power Plant."

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

13. Letter from C. Thomas (NRC) to J. Charnley (GE), dated May 28, 1985.

14. GESTAR II, NEDE-24011, Revision 8, "General Electric Standard Application for Reactor Fuel."

15. Letter from A. Thadani (NRC) to J. Charnley (GE), dated May 12, 1988.

16. Letter from J. Charnley (NRC) to G. Lainas (NRC), "Proposed Amendment 18 to GE Licensing Topical Report NEDO-24011-P-A, dated October 31, 1986.

17. Letter from A. Dromerick (NRC) to P. Siedler (GPU), dated March 21, 1988 (TACs 65138 and 65139).

18. Letter from A. Thadani (NRC) to R. Furia (GPU), dated October 15, 1988.

19. Letter from W. Hodges (NRC) J. Stolz (NRC), dated September 23, 1988.

20. Letter from A. Thadani (NRC) to J. Charnley (GE), dated December 27, 1987.

Dated: October 31, 1988

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TABLE 1

Oyster Creek Cycle 12 Fuel Bundles

	Fuel Type	Average Exposure (GWD/MT)	Number
Irridiated	Exxon VB	17.96	28
	P8DRB239	18.33	112
	P8DRB265	16.30	64
	P8DRB299-7GZ2	10.45	136
	P8DRB299-7GZ1	8.76	48
New	P8DRB-321(EB)	0.0	152
	P8DRB299-7GZ2	0.0	20
Total			560