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

October 27, 1986

Docket No. 50-219

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

Dear Mr. Fiedler:

SUBJECT: OPERATING CYCLE 11 RELOAD (TSCR 149, TAC 61863)

Re: Oyster Creek Nuclear Generating Station

The Commission has issued the enclosed Amendment No. 111 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station. This amendment is in response to your application dated June 17, 1986, as supplemented by letters dated September 17, and October 13, 1986.

This amendment authorizes changes to Section 2.3, Limiting Safety System Settings, and to Section 3.10, Core Limits, of the Appendix A Technical Specifications (TS) to account for the Operating Cycle 11 reload. The changes to Section 2.3 increase (1) the neutron flux scram setting for the average power range monitors (APRM) and (2) the neutron flux control rod block setting. The changes to Section 3.10 increase the minimum critical power ratio (MCPR) limits and revise the maximum allowable average planar linear generation rate (MAPLHGR) for five loop and four loop operation in Figures 3.10-4 and 5, respectively. The changes to the figures replace the MAPLHGR for the existing fuel type P8DRB265L by that for the new fuel type P8DRB299. The MAPLHGR for the existing fuel types P8DRB239 and P8DRB265H in Figures 3.10-4 and 3.10-5 are not being changed by this amendment. Included with these changes are changes to the Bases for TS Sections 2.3 and 3.10.

Although your letter requested that this amendment be effective 30 days after its issuance to accommodate procedural changes, GPU Nuclear (the licensee) stated that the amendment should be issued effective immediately to support Operating Cycle 11. This was by phone call on October 13, 1986. Therefore, this amendment will be effective immediately.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notices.

Sincerely,

Jack N. Donohew, Jr., Project Manager
BWR Project Directorate #1
Division of BWR Licensing

Enclosures:
1. Amendment No. 111to
   License No. DPR-16
2. Safety Evaluation

cc w/enclosures:
See next page

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Docket File	JZwolinski	BGrimes	EButcher	NThompson
NRC PDR	OGC-Beth	ACRS (10)	ELJordan	UCheh
Local PDR	LJHarmon	JDonohew	RDiggs	
BWD#1 Reading	TBarnhart (4)	DVassallo	WJones	
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DBL:PD#1	DBL:PD#1	OGC-BETH	DBL:PD#1
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10/16/86	10/17/86	10/20/86	10/24/86

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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. 111
License No. DPR-16

1. The Nuclear Regulatory Commission (the Commission) has found that:

A. The application for amendment by GPU Nuclear Corporation and Jersey Central Power and Light Company (the licensees) dated June 17, 1986, as supplemented September 17 and October 13, 1986, 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. 111, 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 its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jack N. Donohew, Jr., Project Manager
BWR Project Directorate #1
Division of BWR Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 27, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 111

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
2.3-1	2.3-1
2.3-2	2.3-2
2.3-5	2.3-5
3.10-2	3.10-2
3.10-5	3.10-5
3.10-10	3.10-10
3.10-11	3.10-11

2.3 LIMITING SAFETY SYSTEM SETTINGS

Applicability: Applies to trip settings on automatic protective devices related to variables on which safety limits have been placed.

Objective: To provide automatic corrective action to prevent the safety limits from being exceeded.

Specification: Limiting safety system settings shall be as follows:

FUNCTION	LIMITING SAFETY SYSTEM SETTINGS
A. Neutron Flux, Scram	
A.1 APRM	When the reactor mode switch is in the Run position, the APRM flux scram setting shall be

$$S \leq [(0.90 \times 10^{-6})\, W + 60.8] \left[\frac{FRP}{MFLPD}\right]$$

with a maximum setpoint of 115.7% for core flow equal to 61 x 10^6 lb/hr and greater,

where:

S = setting in percent of rated power
W = recirculation flow (lb/hr)

FRP = fraction of rated thermal power is the ratio of core thermal power to rated thermal power

MFLPD = maximum fraction of limiting power density where the limiting power density for each bundle is the design linear heat generation rate for that bundle.

The ratio of FRP/MFLPD shall be set equal to 1.0 unless the actual operating value is less than 1.0 in which case the actual operating value will be used.

This adjustment may be accomplished by increasing the APRM gain and thus reducing the flow reference APRM High Flux Scram Curve by the reciprocal of the APRM gain change.

A.2 IRM	$\leq$ 38.4 percent of rated neutron flux

FUNCTION	LIMITING SAFETY SYSTEM SETTINGS
B. Neutron Flux, Control Rod Block	The Rod Block setting shall be $S \leq [(0.90 \times 10^{-6}) W + 53.1] [\frac{FRP}{MFLPD}]$ with a maximum setpoint of 108% for core flow equal to 61×10^6 lb/hr and greater. The definitions of S, W, FRP and MFLPD used above for the APRM scram trip apply. The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than 1.0, in which case the actual operating value will be used. This adjustment may be accomplished by increasing the APRM gain and thus reducing the flow referenced APRM rod block curve by the reciprocal of the APRM gain change.
C. Reactor High, Pressure, Scram	$\leq$1060 psig
D. Reactor High Pressure, Relief Valves Initiation	2 @ $\leq$ 1070 psig 3 @ $\leq$ 1090 psig
E. Reactor High Pressure, Isolation Condenser Initiation	$\leq$1060 psig with time delay $\leq$3 seconds
F. Reactor High Pressure, Safety Valve Initiation	4 @ 1212 psig 4 @ 1221 psig $\pm$ 12 psi 4 @ 1230 psig 4 @ 1239 psig
G. Low Pressure Main Steam Line, MSIV Closure	$\geq$825 psig (initiated in IRM range 10)
H. Main Steam Line Isolation Valve Closure, Scram	$\leq$10% Valve Closure from full open
I. Reactor Low Water Level, Scram	$\geq$11'5" above the top of the active fuel as indicated under normal operating conditions
J. Reactor Low-Low Water Level, Main Steam Line Isolation Valve Closure	$\geq$7'2" above the top of the active fuel as indicated under normal operating conditions

APRM scram. The transients requiring a scram by nuclear instrumentation are the loss of feedwater heating and the improper startup of an idle recirculation loop. The loss of feedwater heating transient is not affected by the range 10 IRM since the feedwater heaters will not be put into service until after the LPRM downscales have cleared, thus insuring the operability of the APRM system. This will be administratively controlled. The improper startup of an idle recirculation loop becomes less severe at lower power level and the IRM scram would be adequate to terminate the flux excursion.

The Rod Worth Minimizer is not required beyond 10% of rated power. The ability of the IRMs to terminate a rod withdrawal transient is limited due to the number and location of IRM detectors. An evaluation was performed that showed by maintaining a minimum recirculation flow of 39.65×10^6 lb/hr in range 10 a complete rod withdrawal initiated at 35% of rated power or less would not result in violating the fuel cladding safety limit. Therefore, a rod block on the IRMs at less than 35% of rated power would be adequate protection against a rod withdrawal transient.

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent gross rod withdrawal at constant recirculation flow rate to protect against grossly exceeding the MCPR Fuel Cladding Integrity Safety Limit. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the safety limit increases as the flow decreases for the specified trip setting versus flow relationship. Therefore, the worst-case MCPR, which could occur during steady-state operation, is at 108% of the rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the incore LPRM system. As with APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum fraction of limiting power density exceeds the fraction of the rated power, thus preserving the APRM rod block safety margin. As with the scram setting, this may be accomplished by adjusting the APRM gains.

The settings on the reactor high pressure scram, anticipatory scrams, reactor coolant system relief valves and isolation condenser have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. In addition, the APRM neutron flux scram and the turbine bypass system also provide protection for these safety limits, e.g., turbine trip and loss of electrical load transients (5). In addition to preventing power operation above 1060 psig, the pressure scram backs up the other scrams for these transients and other steam line isolation type transients. Actuation of the isolation condenser during these transients removes the reactor decay heat without further loss of reactor coolant thus protecting the reactor water level safety limit.

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

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 initated 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 (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.	1.45
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).	1.45
3.	All B, C, and D LPRM inputs to the APRM system are operating and no APRM channels are inoperable or bypassed.	1.45

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 fuel-type P8x8R 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% rated 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 results for the reference cycle (3) indicate that the steady state MCPR required to protect the minimum transient MCPR of 1.07 is 1.23 or higher for the worst case APRM status condition (APRM STATUS 1). This steady state limit conservatively applies to APRM status 2 and 3. The steady state MCPR values for APRM status conditions 1, 2, and 3 will be evaluated each cycle. In order to provide for a limit which is considered to be bounding to future operating cycles, the limits for each APRM status condition have been conservatively adjusted upward to 1.45. This is also the assumed value for LOCA analysis.

The time interval of eight (8) hours to adjust the steady state 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 ΔCPR 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.

FIGURE 3.10-4

MAXIMUM ALLOWABLE AVERAGE PLANAR
LINEAR HEAT GENERATION RATE
(FIVE LOOP OPERATION)

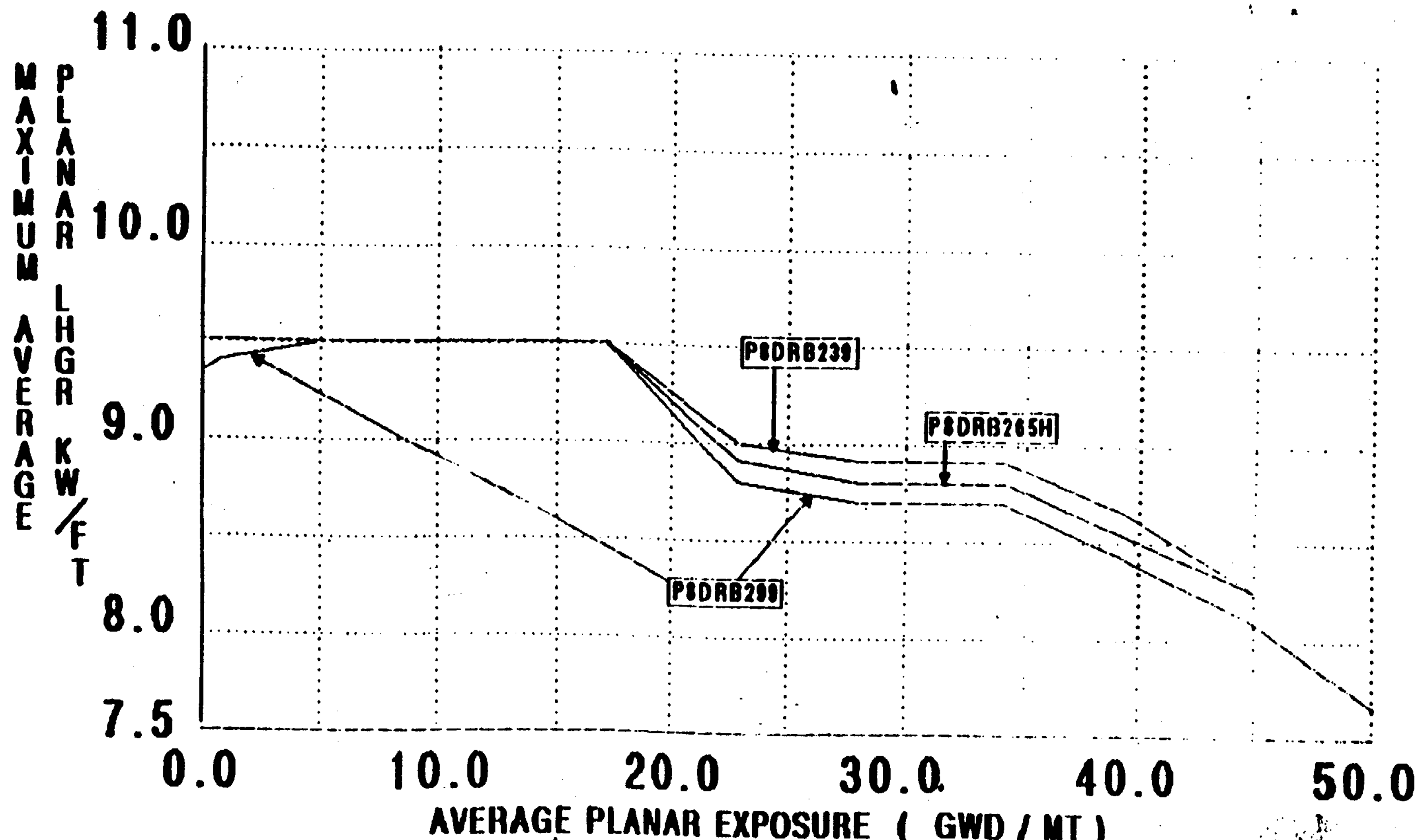
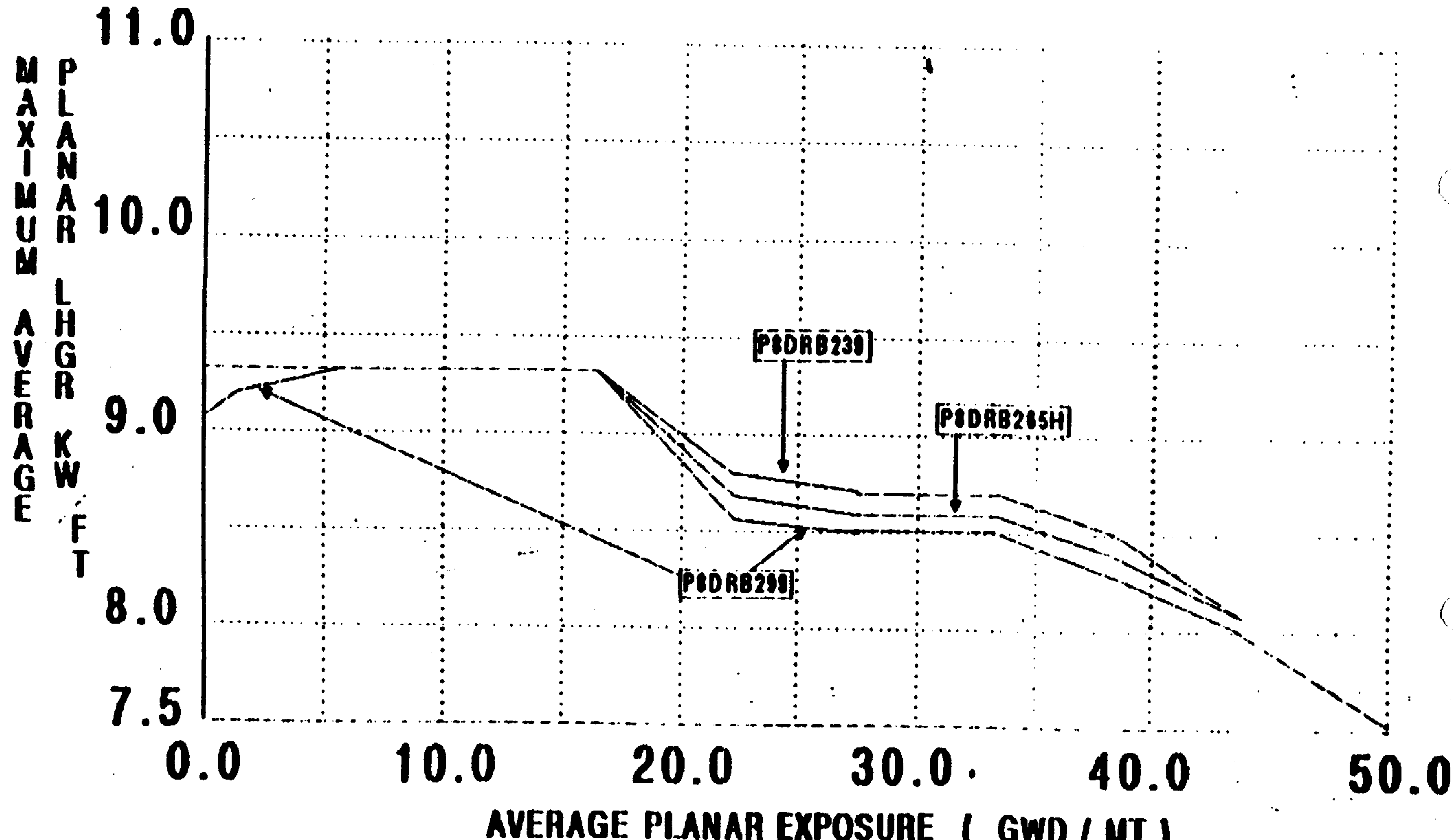


FIGURE 3.10-5

MAXIMUM ALLOWABLE AVERAGE PLANAR
LINEAR HEAT GENERATION RATE
(FOUR LOOP OPERATION)



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 111 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 letters dated June 17, September 17, and October 13, 1986, GPU Nuclear (the licensee) requested an amendment to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station (Oyster Creek). This amendment would authorize changes to Section 2.3, Limiting Safety System Settings, and to Section 3.10, Core Limits, of the Appendix A Technical Specifications (TS) to account for the Operating Cycle 11 reload.

The changes to Section 2.3 would increase (1) the neutron flux scram setting for the average power range monitors (APRM) and (2) the neutron flux control rod block setting. The changes to Section 3.10 would increase the minimum critical power ratio (MCPR) limits and revise the maximum allowable average planar linear heat generation rate (MAPLHGR) for five loop and four loop operation in Figures 3.10-4 and 5, respectively. The changes to the figures would replace the MAPLHGR for the existing fuel type P8DRB256L by that for the new fuel type P8DRB299. The MAPLHGR for the existing fuel types P8DRB239 and P8DRB265H in Figures 3.10-4 and 3.10-5 are not being changed by this amendment. Included with these changes are changes to the Bases for TS Sections 2.3 and 3.10.

2.0 DISCUSSION

By letter dated June 17, 1986 (Reference 1), the licensee proposed to change the TS in the areas of the APRM Scram and Rod Block Lines and the MCPR and MAPLHGR limits for Oyster Creek to accommodate the Operating Cycle 11 reload. The submittal references the staff-reviewed and approved NEDO-24195, "General Electric Reload Fuel Application for Oyster Creek" which provides the bases for the TS changes necessary for the Cycle 11 operation and the attached Appendix D (to NEDO-24195) which is the summary of the results of the Cycle 11 reload core design and safety analysis.

The Cycle 11 core will retain 372 irradiated fuel assemblies of Exxon Type VB, and GE types P8DRB239 and P8DRB265H from the Cycle 10 and will add 188 fresh fuel assemblies (about 34% of the fuel) of GE Types P8DRB265H, P7DRB299LA and P8DRB299H (References 1, 2, 3). The reload is based on a Cycle 10 exposure of 15.769 GWD/t and the loading will be a conventional scatter pattern with low reactivity fuel on the periphery.

The safety evaluation for the Cycle 11 reload included staff comparison of NEDO-24195, "General Electric Reload Fuel Application for Oyster Creek" with NEDE-24011, "General Electric Boiling Water Reactor Generic Reload Fuel Application" which was previously reviewed and approved by the staff for reference in the safety analysis of the GE Boiling Water Reactors. The staff concluded that the methodology and procedures employed in the reload design and analysis are essentially the same as those described in the previously approved NEDO-24011 and are acceptable. The procedures used to establish operating limits are similar to those previously approved and are acceptable. The safety analyses performed in support of the Cycle 11 core design use the methods described in the NEDO-24195. MAPLHGR values provided for the new reload fuel assemblies of P8DRB265H and P8DRB299 are provided based on LOCA analyses using approved methodologies and parameters. (Reference 4)

The projected End of Cycle 11 Maximum Batch Average Exposure was 27600 MWD/MTU. This GE fuel design has been previously approved and operated well beyond this burnup range. A summary of the results of the Cycle 11 reload core design and safety analyses are given in Appendix D of the Cycle 11 Reload Submittal.

In the core-related areas of fuel design, thermal-hydraulic design, nuclear design and safety analyses of the postulated accidents and transients, the licensee has relied on the results presented in the approved GE topical report NEDE-24011, "General Electric Standard Application for Reactor Fuel (GESTAR II)," (Reference 4). In addition, the licensee submitted a supplemental reload licensing document (Reference 3) which provides the results of other analyses necessary to justify Cycle 11 operation but which are not included in GESTAR II.

3.0 EVALUATION

3.1 Fuel Design

A total of 188 fresh GE type fuel assemblies of P8DRB65H, P8DRB299LA and P8DRB299H (References 1, 2, 3), which are pressurized 8x8 retrofit barrier fuel assemblies will be loaded for the Cycle 11 operation. Since the new pressurized 8x8 retrofit barrier fuel has been previously staff-reviewed and approved (Reference 5), we conclude that the fuel assemblies are acceptable for the Cycle 11 operation. The new fuel assemblies will reside with 372 irradiated 8x8 fuel assemblies of prior Exxon and GE designs presently in the core. The fuel designs for the irradiated assemblies of Exxon Type VB, and GE Types P8DRB239 and P8DRB265H have been previously approved and operated. (References 4, 5)

3.2 Nuclear Design

The nuclear design and analysis of the proposed reload has been performed by the methods described in Reference 5. Reference 5 has been approved for use in the design and analysis of reloads in BWR reactors and its use is acceptable for this reload. We have reviewed the results of the nuclear design analysis for the Oyster Creek Cycle 11 and have determined that they are acceptable since the nuclear parameters are within the range of those normally obtained for similar cores and were obtained with acceptable methods.

3.3 Thermal-Hydraulic Design

The objective of the review of the thermal-hydraulic design of the Core Cycle 11 operation is to confirm that the thermal-hydraulic design has been accomplished using acceptable methods, and to assure an acceptable margin of safety from conditions which could lead to fuel damage during normal operation and anticipated transients and to assure that the core is not susceptible to thermal-hydraulic instability.

The review included the following areas:

(1) Minimum Critical Power Ratio (MCPR) and the related changes to the TS.

(2) Maximum Average Planar Linear Heat Generating Rate (MAPLHGR) and the related changes to the TS.

A safety limit value of MCPR is imposed to assure that 99.9 percent of the fuel rods in the core will not experience boiling transition during normal operation and anticipated operational transients. As stated in Reference 2, the approved safety limit MCPR for the Oyster Creek P8x8R reload core is 1.07. The safety limit of 1.07 was used for the Cycle 11 analysis.

The licensee has proposed that two MAPLHGR curves for the fresh fuel bundles of P8DRB299 be added to the Oyster Creek Technical Specifications to replace a reference to the same curves in a proprietary General Electric Topical Report (Reference 4). This is an administrative change which we find to be acceptable and appropriate.

3.4 Transient and Accident Analyses

The corewide transient and accident analyses for Turbine Trip without Bypass, Loss of 100°F Feedwater Heating, Feedwater Controller Failure, MSIV Closure and Rod Withdrawal Errors (RWE) were performed using approved methods described in Reference 4 and the results of the accident analyses are acceptable for Cycle 11. (Appendix D to NEDO-24195)

The Turbine Trip Without Bypass was the most limiting transient for Cycle 11 with a maximum MCPR of 1.41 as calculated by ODYN option A and 1.36 by ODYN Option B. This compares to 1.40 for Cycle 10 where the most limiting transient was the RWE (Rod Withdrawal Error) case. The licensee has conservatively selected an operating limit MCPR value of 1.45. This will not create any operating difficulties since Cycle 11 is expected to operate with an MCPR margin of 20% or greater.

3.5 Thermal-Hydraulic Stability

The assurance that the reactor is stable and has adequate stability design margin is demonstrated analytically when a core stability decay ratio less than 1.0 is calculated using approved methods. For Oyster Creek Cycle 10, a limiting stability decay ratio of 0.67 (or 0.87 including approved uncertainty values for the calculation method) was calculated (Reference 9). This was not reanalyzed for the Cycle 11 Core. However, operating conditions for Cycle 11 are essentially the same as for Cycle 10. Changes in stability margin due to the difference in core characterisitics are small. The licensee has concluded that there is sufficient margin to assure thermal-hydraulic stability for Cycle 11 (References 7, 8, 9). We find this acceptable for Oyster Creek (a BWR 2) in accordance with Generic Letter 86-02 (Reference 8).

3.6 Technical Specifications Change

The proposed revision of the APRM Scram and Rod Block Lines to provide greater flexibility during startup and power escalation to rated conditions was reviewed. The staff determined that the revision is acceptable since the methodology and procedures used are staff-reviewed and approved in References 3 and 4.

The MCPR Limits are revised from 1.40 to 1.45 and maximum allowable average planar LHGR curves for 5 and 4 loop operations as described in Appendix D of Reference 1 are added. The staff has reviewed the proposed TS changes for Cycle 11 and concludes that they are acceptable.

3.7 Extended Burnup Evaluation

In response to a staff request, the licensee provided information that the Projected End of Cycle 11 Maximum Assembly Exposure is 29300 MWD/MTU (Reference 9). This GE fuel design has been previously approved and operated well beyond this burnup range (Reference 5). Thus, extended burnup is not a factor in operation of the Cycle 11 core.

3.8 Fuel Performance

In Licensee Event Report (LER) No. 86-016, dated July 30, 1986, the licensee reported fuel clad failures associated with 47 fuel bundles during Cycle 10 operation. The offgas radiation level continually increased during the cycle (from 50,100 mCi/sec to 224,000 mCi/sec) and the I-131/I-133 fission product ratio also increased (from .069 to .144). All leaking fuel bundles and all other fuel bundles in the same control cell were removed from the Cycle 11 reload. Nearly all of the failures (45 of 47) occurred in the same EXXON fuel batch. Although the licensee's investigation into the cause of the fuel cladding failures is incomplete, it is believed that the failure mechanism involved defective cladding or pellet/clad interaction, possibly aggravated by failure to follow the fuel preconditioning recommendations of the fuel supplier.

Concerns regarding the fuel failures and the progress of the licensee's investigation were reviewed by the NRC regional office as part of a safety inspection (Reference 10). The licensee identified several actions intended to preclude the repetition of excessive fuel failures during Cycle 11, including:

(1) the acquisition of a new load line limit computer analysis which will permit maneuvering at high power by adjusting recirculation pump speed to minimize the contribution of rod position changes to pellet/clad interaction;

(2) revision of the licensee's core monitoring computer program to eliminate errors which had made the program ineffective for conformance to fuel preconditioning operating recommendations;

(3) the addition of a new computer program to track and trend the weekly offgas reactivity levels and the reactor coolant fission product ratios to aid in early identification of fuel failures during Cycle 11 operation.

In addition, the licensee has committed to keep the NRC informed of the fuel failure investigation results.

The staff concludes that the licensee is acting prudently to reduce the probability of fuel failures and to minimize the activity release in the offgas during Cycle 11 operation, and we find this acceptable. We will continue to follow the investigation of the Cycle 10 fuel failures and will pursue any additional actions that may be indicated with the licensee.

3.9 Conclusion

The staff concludes, as discussed above, that acceptable methods and procedures were used to perform the design and analysis of the Oyster Creek reactor reload for Cycle 11 operation and that the the licensee's proposed amendment is correctly based on the results of that design and analysis. Therefore, the staff concludes that this amendment is acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to 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, this 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 nor environmental assessment need be prepared in connection with the issuance of this amendment.

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 the issuance of this 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, P. B. Fiedler (GPUN) to J. A. Zwolinski (NRC), Oyster Creek Nuclear Generating Station Docket No. 50-219 Technical Specification Change Request (TSCR) No. 149, dated June 17, 1986.

2. W. H. Hetzel (Oyster Creek) to R. B. Lee (GPUN), "Oyster Creek Bundle Name Changes," dated February 28, 1986.

3. Letter, Peter B. Fiedler (Oyster Creek) to John A. Zwolinski, Oyster Creek Nuclear Generating Station Docket No. 50-219, Technical Specification Change Request (TSCR) No. 149, Revision 1 of Appendix D to NEDO-24195, dated September 17, 1986.

4. "General Electric Reload Fuel Application for Oyster Creek," NEDO-24195 79NED288, Class I, August, 1979.

5. GESTAR II, General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A-6-US Class III, April, 1983, and its Proposed Amendment 13 (to Revision 6 to NEDE-24011-P-A) submitted September 24, 1985 and approved March 26, 1986.

6. Telecopy from J. Lachermayer (GPUN) to J. Donohew (NRC), "Projected End of Cycle 11 Fuel Exposures," dated July 29, 1986.

7. General Electric Service Information letter No. 380, Revision 1, February 10, 1984.

8. Generic Letter No. 86-02, "Technical Resolution of Generic Issue B-19-Thermal Hydraulic Stability," January 23, 1986.

9. Letter, R. F. Wilson (GPUN) to John A. Zwolinski, Oyster Creek Nuclear Generating Station Docket No. 50-219, Technical Specification Change Request (TSCR) No. 149, dated October 13, 1986.

10. NRC Region I Inspection Report Number 50-219/86-25 (on 4.0 Fuel Clad Failures) dated September 18, 1986.

Principal Contributor: U. Cheh

Dated: October 27, 1986