August 26, 199

Mr. Michael B. Roche
Vice President and Director
GPU Nuclear Corporation
Oyster Creek Nuclear Generating Station
P.O. Box 388
Forked River, NJ 08731

SUBJECT: OYSTER CREEK - ISSUANCE OF AMENDMENT RE: CHANGE IN THE SAFETY LIMIT MINIMUM CRITICAL POWER RATIO (TAC NO. M96722)

Dear Mr. Roche:

The Commission has issued the enclosed Amendment No. 192 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated October 4, 1996, and supplemented by letters dated June 10 and August 15, 1997.

The amendment reflects a change in the Safety Limit Minimum Critical Power Ratio (SLMCPR) and as a result, a change in the operating Minimum Critical Power Ratio (MCPR).

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

Sincerely,

Original signed by

Ronald B. Eaton, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-219

Enclosures: 1. Amendment No. 192 to DPR-16
2. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION
See attached list

DFOl

DOCUMENT NAME: G:\EATON\OC.AMD
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	PM:PD1-3	E	LA:PD3-1	E	OGC		DD:DRPE	
NAME	REaton:		CJamerson		MYoung		JZwolinski	
DATE	8/20/97		8/20/97		8/25/97		8/26/97	

OFFICIAL RECORD COPY

DATED: August 26, 1997

AMENDMENT NO. 192 TO FACILITY OPERATING LICENSE NO. DPR-16-OYSTER CREEK

DISTRIBUTION:
Docket File
PUBLIC
PDI-3 RF
BBoger
REaton
CJamerson
OGC
GHill (2)
ACRS
WBeckner
CHehl
THarris (TLH3)

M. Roche
GPU Nuclear Corporation

Oyster Creek Nuclear
Generating Station

cc:

Ernest L. Blake, Jr., Esquire
Shaw, Pittman, Potts & Trowbridge
2300 N Street, NW
Washington, DC 20037

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

BWR Licensing Manager
GPU Nuclear Corporation
1 Upper Pond Road
Parsippany, NJ 07054

Mayor
Lacey Township
818 West Lacey Road
Forked River, NJ 08731

Licensing Manager
Oyster Creek Nuclear Generating Station
Mail Stop: Site Emergency Bldg.
P.O. Box 388
Forked River, NJ 08731

Resident Inspector
c/o U.S. Nuclear Regulatory Commission
P.O. Box 445
Forked River, NJ 08731

Kent Tosch, Chief
New Jersey Department of
Environmental Protection
Bureau of Nuclear Engineering
CN 415
Trenton, NJ 08625

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

GPU NUCLEAR CORPORATION

AND

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 192
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 October 4, 1996, as supplemented June 10, and August 15, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 192 , are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John A. Zwolinski, Deputy Director
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 26, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 192

FACILITY OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace the following pages of the Appendix A, Technical Specifications, with the attached pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
2.1-1	2.1-1
2.1-2	2.1-2
2.1-3	2.1-3
3.10-1	3.10-1
3.10-2	3.10-2
3.10-3	3.10-3
3.10-4	3.10-4

SECTION 2

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT - FUEL CLADDING INTEGRITY

Applicability: Applies to the interrelated variables associated with fuel thermal behavior.

Objective: To establish limits on the important thermal hydraulic variables to assure the integrity of the fuel cladding.

Specifications:

A. When the reactor pressure is greater than or equal to 800 psia and the core flow is greater than or equal to 10% of rated, the existence of a minimum CRITICAL POWER RATIO (MCPR) less than 1.09* shall constitute violation of the fuel cladding integrity safety limit.

B. When the reactor pressure is less than 800 psia or the core flow is less than 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

C. In the event that reactor parameters exceed the limiting safety system settings in Specification 2.3 and a reactor scram is not initiated by the associated protective instrumentation, the reactor shall be brought to, and remain in, the COLD SHUTDOWN CONDITION until an analysis is performed to determine whether the safety limit established in Specification 2.1.A and 2.1.B was exceeded.

D. During all modes of reactor operation with irradiated fuel in the reactor vessel, the water level shall not be less than 4'8" above the TOP OF ACTIVE FUEL.

Bases:

The fuel cladding integrity safety limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedure used to calculate the

* Applicable for cycle 16 only.

critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity safety limit is defined as the CRITICAL POWER RATIO in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB[1], which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) - Boiling Length (L), GEXL, correlation.

The use of the GEXL correlation is not valid for the critical power calculations at pressures below 800 psia or core flows less than 10% of rated. Therefore, the fuel cladding integrity safety limit is protected by limiting the core thermal power.

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psi or core flow less than 10% is conservative.

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 2.1.A or 2.1.B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. Specification 2.1.C requires that appropriate analysis be performed to verify that backup protective instrumentation has prevented exceeding the fuel cladding integrity safety limit prior to resumption of POWER OPERATION. The concept of not approaching a Safety Limit provided scram signals are OPERABLE is supported by the extensive plant safety analysis.

If reactor water level should drop below the TOP OF ACTIVE FUEL, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. With a water level above the TOP OF ACTIVE FUEL, adequate cooling is maintained and the decay heat can easily be accommodated. It should be noted that during power generation there is no clearly defined water level inside the shroud and what actually exists is a mixture level. This mixture begins within the active fuel region and extends up through the moisture separators. For the purpose of this specification water level is defined to include mixture level during power operations.

The lowest point at which the water level can presently be monitored is 4'8" above the TOP OF ACTIVE FUEL. Although the lowest reactor water level limit which ensures adequate core cooling is the TOP OF ACTIVE FUEL, the safety limit has been conservatively established at 4'8" above the TOP OF ACTIVE FUEL.

REFERENCES

(1) NEDE-24011-P-A-11, General Electric Standard Application for Reactor Fuel and US Supplement NEDE-24011-P-A-11-US.

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.

The MCPR limit for each cycle as identified in the COLR shall be greater than or equal to 1.49.

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 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 determine 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 the average linear heat generation rate 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 provided 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.

A lower bound of 1.49 has been established for the operating limit MCPR value to provide sufficient margin to the MCPR safety limit in the event of reactor thermal-hydraulic instability. The 1.49 limit will be considered against the minimum operating CPR limit based on reload transient and accident analysis. The higher of core stability or reactor transient and accident determined MCPR will be used to determine the cycle operating limit.

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.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 192

TO FACILITY OPERATING LICENSE NO. DPR-16

GPU NUCLEAR CORPORATION AND

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated October 4, 1996, as supplemented June 10, 1997, the GPU Nuclear Corporation (the licensee) requested an amendment to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station (OCNGS). The requested changes would revise the Safety Limit Minimum Critical Power Ratio (SLMCPR). The supplemental letter provided clarifying information within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The licensee requested a change to the OCNGS Cycle 16 TSs in accordance with the 10 CFR 50.91(b)(1). The proposed revisions to TS 2.1.A and 3.10.C are described below.

2.1 The licensee proposed to change the safety limit MCPR in TS 2.1.A from 1.07 to 1.09 when the reactor pressure is $\geq$ 800 psia and the core flow is $\geq$ 10% of rated based on the cycle-specific analysis performed by General Electric Company (GE) for OCNGS Cycle 16 mixed core of GE8B/GE9B fuel (all GE 8x8 fuel). It is also proposed to replace the reference NEDO-24195 with NEDE-24011-P-A-11 [proprietary information not publicly available]. The cycle-specific parameters were used including the actual core loading, the most limiting permissible control blade patterns, actual bundle parameters, and the full cycle exposure range.

The staff has reviewed the proposed TS changes which are based on the analyses performed using OCNGS cycle-specific inputs and approved methodologies including GESTAR II (NEDE-24011-P-A-11, Sections 1.1.5 and 1.2.5) and found acceptable since the cycle-specific analysis is conservative compared with the generic GE9B SLMCPR evaluation and due to (1) the OCNGS Cycle 16 is not an equilibrium core, (2) the OCNGS Cycle 16 analysis produces both a flatter bundle-by-bundle and pin-by-pin power distribution than that used to perform the GE9B SLMCPR evaluation, and (3) the OCNGS Cycle 16 is loaded with a higher latest reload batch fraction and a higher latest reload average weight percent enrichment. Use of this methodology ensures that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated, thereby preserving fuel cladding integrity.

2.2 The licensee proposed to change the MCPR limit in TS 3.10.C from 1.47 to 1.49 in order to reflect the revision to the SLMCPR in TS 2.1.A.

Based on our review, we conclude that the changes to the TSs and their associated Bases are acceptable for OCNGS Cycle 16 application since the changes are analyzed based on the NRC-approved method and ensure that 99.9% of the core will avoid transition boiling.

2.3 The licensee also requested changes in capitalization for certain definitions that appear in the above specifications and bases and a change to provide for a uniform type font for Sections 2.1 and 3.10. These changes are typographical, provide clarity to the TSs, and are acceptable to the staff.

In addition, on TS page 3.10-2 the staff discovered a typographical error in the second paragraph of the Bases and after confirming the correct usage with the licensee, the word "determined" was corrected to "determine."

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC 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 the amendment involves no significant hazards consideration, and there has been no public comment on such finding (61 FR 57484). 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 the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: T. Huang

Date: August 26, 1997