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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 75 TO PROVISIONAL OPERATING LICENSE NO. DPR-16

GPU NUCLEAR CORPORATION AND

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated April 21, 1980 as supplemented March 9, 1981, August 31, 1982, and October 28, 1983, GPU Nuclear Corporation (GPU) submitted a request for changes to the Oyster Creek Nuclear Generating Station's Provisional Operating License No. DPR-16 Technical Specification (TS) to accommodate the Cycle 10 reload. The submittal included NEDO-24195, "General Electric Reload Fuel Application for Oyster Creek" which provides the basis for the TS changes necessary for Core 10 operation. The licensee also provided submittals of July 22, 1983 which amended NEDO-24195, and May 1, 1984 which responded to the staff's requests for additional information.

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on July 20, 1983 (48 FR 33081) and July 20, 1984 (49 FR 29495). No request for hearing or public comments were received.

2.0 DISCUSSION AND EVALUATION

2.1 Methodology Topical

2.1.1 Description of Report

NEDO-24195 describes General Electric (GE) supplied reload fuel mechanical design, nuclear evaluation methods, steady-state thermal-hydraulic methods, and reactor limits determination. In addition Appendix A presented the format in which the results of the reload analyses were submitted to the staff for review. These are essentially identical to those described in the previously approved [Letter to Gridley (GE) from Eisenhut (NRC) dated May 12, 1978] report NEDE-24011-PA, "GE Boiling Water Reactor Generic Reload Fuel Application," except for those features and considerations unique to the Oyster Creek reactor.

2.1.2 Fuel Mechanical Design

Section 2 of NEDO-24195, "Fuel Mechanical Design," describes the design bases, limits, and evaluation of the thermal, mechanical, and materials design of the Oyster Creek Fuel System. The fresh fuel assemblies introduced in the Cycle 10 reload are General Electric Prepressurized 8x8 Retrofit (P8x8R) assemblies with lattice and bundle average enrichments described in Section 1 of NEDO-24195.

As stated previously, the Reload Application for Oyster Creek (NEDO-24195) generally follows the format used in the General Electric Reload Fuel Application (NEDE-24011), which has been reviewed and approved by the NRC. Section 2 in both reports are identical except for the deletion of proprietary information from the Oyster Creek report. Because no significant differences exist for the GE P8x8R fuel design in the Oyster Creek application, the staff finds that previous approval of Section 2 in NEDE-24011 is equally applicable to the corresponding section of NEDO-24195.

It should be noted that several amendments to NEDE-24011 are currently under staff review. Their approval is expected to bring the most recent version of NEDE-24011, now called the General Electric Standard Application for Reactor Fuel (GESTAR-II), into conformance with the NRC Standard Review Plan, resolve several technical issues raised since the original approval of NEDE-24011, and allow the application of a number of recently approved analytical methods. In those areas where issues have been raised since the original approval of NEDE-24011 (e.g., fission gas release at high burnup), the Oyster Creek application has been modified (e.g., NEDO-24195, Amendment 6, Section 5.5.2) in a manner similar to that used in the GE analysis of other operating reactors. The staff finds this acceptable. Because the Oyster Creek Application (NEDO-24195) refers to the revision of NEDE-24011 which is approved by the date a specific analysis is initiated, final resolution of any technical issue of GESTAR-II will eventually (and automatically) apply to Oyster Creek. The staff finds this method of reference to future revisions of NEDE-24011 acceptable as well.

2.1.3 Nuclear Evaluation Methods

Section 3 of NEDO-24195, "Nuclear Evaluation Methods," describes the techniques used to obtain the nuclear parameters of the fuel bundles and the core. Reference is made to approved topical reports which give detailed descriptions of the methods and their verification by comparison with measurements. In addition the procedures used to determine the reference loading pattern for the new cycle are discussed, along with procedures for "fine-tuning" the loading to account for unexpected events (e.g., end-of-cycle coastdown in the current cycle). These procedures are essentially identical to those in the approved generic reload design report, and are acceptable.

2.1.4 Steady-State Hydraulic Models

Section 4 of NEDO-24195, "Steady-State Hydraulic Models," describes the methods used to obtain the pressure drops and flow distribution for the steady-state thermal-hydraulic analyses of the core. The present Exxon fuel bundles are modeled by using the characteristics of General Electric bundles which are similar as regards to fuel rod geometry and enrichment. The core model requires hydraulic descriptions of orifices, lower tieplates, fuel rods, fuel rod spacers, upper tieplates, the fuel channel and core bypass flow paths.

The flow distribution to the fuel assemblies and bypass flow paths is calculated on the assumption that the pressure drop across all fuel assemblies and bypass flow paths is the same. This has been confirmed by in-plant measurements as cited by references. The components of bundle pressure drop considered are friction, locale, elevation and acceleration, each of which is discussed in separate subsections which include the applicable equations and references. The equations given for the various pressure drops were checked and found to be in agreement with accepted practice. A subsection on bypass flow indicates that full scale tests have been performed to compare against independent analytical models which predicted the test results. One flow path where a significant amount of bypass flow can occur is due to channel wall deflection at the lower tieplate. To provide control over this flow path, optional finger spring seals can be added to most reload fuel assemblies. This can reduce bypass flow over a wide range of channel wall deflections. The analysis method provides for modeling fuel assemblies with and without finger seals by supplying separate hydraulic constants to represent both finger spring seal and non-finger spring seal bundles for proper calculation. Though the by-pass flow paths considered in the analyses are shown schematically, no details are given to show that the analyses correctly model this flow. However, the steady-state hydraulic models use procedures which are essentially the same as those in the approved generic reload design report, NEDE-24011-P-A-1, and are therefore acceptable for use in the Oyster Creek reload fuel application report.

2.1.5 Reactor Limits Determination

The primary concern addressed in Section 5 of NEDO-24195 is to maintain nucleate boiling on at least 99.9% of the fuel rods during normal operation and moderate frequency transient events.

The measure of transition from nucleate to film boiling used in BWR reactor thermal hydraulics is the Critical Power Ratio (CPR). A CPR of unity implies transition to film boiling; less than unity implies film boiling which could lead to fuel damage. The CPR is dependent upon a number of operating parameters such as core flow, core power, and reactor pressure. These are subject to change during normal operation and have certain measurement uncertainties. Statistical analyses are conducted which take into account these uncertainties and operational variations to determine the CPR that meets the 99.9% criteria for normal operation; this value of CPR is called the safety limit minimum critical power ratio (safety limit MCPR). The safety limit MCPR protects the core during normal operation. However, a higher value of CPR, the operational limit of transient MCPR, must be maintained to prevent violation of the 99.9% requirement during transients of moderate frequency.

Section 5 of NEDO-24195 describes procedures for setting operating limits for the cycle and for setting certain limits (e.g., APRM rod blocks) in the protection system. These limits are established so that violation of fuel thermal limits does not occur for normal operation or anticipated operational occurrences and that acceptable consequences are calculated for accidents. Limits on the core-wide value of MCPR and clad strain are established. The various anticipated transients are then examined for their effect on MCPR and linear heat generation rate. The change in these quantities during the transients is then combined with the safety limit to obtain required operating limits for each transient. The most limiting of these limits then becomes the Technical Specification value for the parameter (e.g., MCPR).

2.1.5.1 <u>Reactor Limits Determination Overview</u>

Section 5.0 presents an overview which addresses the determination of reactor operating limits required to meet safety requirements. The safety limit applicable to this review is that of selecting values of CPR which will provide an adequate safety factor during normal plant operations and during moderate frequency transients. Two separate entities are used in this report--both of which are called MCPR. The first is the safety limit MCPR which will ensure that 99.9% of the fuel rods are in nucleate boiling. This value of CPR must exceed unity by a certain margin to take into account fluctuations in plant operating conditions and measurement and computational uncertainties in plant operating parameters. This margin is determined by conducting statistical analyses which model the plant and take into account the range of variation and measurement uncertainties of the parameters which affect CPR. General Electric found that a value of safety limit MCPR of 1.07 is sufficient. Section 5.1 deals with determining this quantity. The second value of MCPR of interest is the operating limit MCPR. The operating limit MCPR is the value of MCPR at which the plant must be operated to ensure that the MCPR will not fall below the safety limit MCPR during a transient. This value is obtained by determining the largest drop in critical power ratio found in any of the analyzed transients and adding that to the normal MCPR. The analyses showed that Oyster Creek should be operated at a CPR at or above 1.25 with the General Electric reload. Section 5.2 discusses the operating limit MCPR. Section 5.4 describes the analyses used to show that the reactor will operate stably with the reload and Section 5.5 describes the accident evaluation methodology.

Section 5.0 is essentially identical to the staff approved report, NEDE-24011-P-A-1, and is therefore acceptable.

2.1.6 Fuel Cladding Integrity Safety Limit

Section 5.1 describes how the normal MCPR is obtained. This section is very similar to NEDE-24011-P-A-1 but contained several items requiring clarification or additional information. These include:

(a) the validity of the modeling of the Exxon bundles (which make up 80% of the reload core) by using fuel which most closely matches the thermal-hydraulic and nuclear characteristics of the Exxon fuel.

(b) the statement that the "large reload core analysis results conservatively apply to Oyster Creek for all General Electric-supplied reload cycles."

(c) that the General Electric bounding analysis is conservative although some of the plant-unique uncertainties may be greater for Oyster Creek.

In a letter dated May 1, 1984 GPU provided additional information demonstrating that the Exxon bundles can be modeled by General Electric bundles. General Electric stated that NEDO-24195 will be updated by including the results of an ODYN code analysis for Oyster Creek. This analysis uses a thermal-hydraulic model for Exxon fuel and adequately resolves these issues.

2.1.7 MCPR Operating Limit Calculational Procedure

Section 5.2 describes the procedure for obtaining the operating limit MCPR, the MCPR which will keep 99.9% of the fuel rods in nucleate boiling during moderate frequency transients. Topics which are discussed include: (a) the system model used; (b) nuclear considerations (scram and void reactivity, Doppler coefficient); (c) inputs to transient analyses; (d) need to increase the operational limit MCPR at low flow; (e) transients chosen for analysis; (f) description of the analysis of each of those four transients; (g) exposure-dependent limits; and (h) effect of fuel densification on MCPR operating limit.

This section closely parallels the same section found in NEDE-24011-P-A-1 and was acceptable after a few items were clarified. This included questions on whether or not several revisions to the subject report and several references to the reference report were NRC approved. Also the licensee submitted a response related to the margin between safety valve setpoints and peak transient pressures and is acceptable.

2.1.8 Stability Analysis Method

Section 5.4 is identical to NEDE-24011-P-A-1 except for two revisions dated September 1979 and one revision dated January 1980. These revisions were found acceptable by the staff.

2.1.9 Accident Evaluation Methodology

Various accidents are analyzed in Section 5.5 to determine the operating limits (initial conditions) which preclude unacceptable consequences. For example, operating limits for the average planar heat generation rate (APLHGR) are established so that peak clad temperatures are not exceeded in the loss-of-coolant accident.

The following transients and accidents analyses are described.

Transients

- turbine trip without bypass
- generator load rejection without bypass
- loss of feedwater heating
- feedwater controller failure
- rod withdrawal events

Accidents

- o control rod drop
- o loss-of-coolant accident
- o main steamline break
- o fuel misloading
- o recirculation pump seisure
- o refueling accident (assembly drop)

In addition, the establishment of the core MCPR safety limit and the analysis of core thermal-hydraulic stability are discussed. The discussions are essentially the same as those in the approved generic topical.

2.2 Summary of Evaluation

The following discussion summarizes the evaluation of NEDO-24195 by the staff.

2.2.1 Fuel Mechanical Design

The fuel thermal, mechanical and materials design methods employed for Oyster Creek reloads are described in Section 2 of NEDO-24195. Section 2 of that report is essentially the same as the corresponding section of NEDO-24011, which has been reviewed and approved by the NRC for reference in the safety analysis of other boiling water reactors. Because the procedures employed in the reload design and analysis are identical (as far as the fuel design is concerned), the staff finds the Oyster Creek application acceptable.

2.2.2 Nuclear Design

The nuclear design methods employed for Oyster Creek reloads are described in the various topical reports referenced in Section 3 of the report. These reports have been reviewed and approved by the staff for reference to design methods for boiling water reactors. They are therefore acceptable for use for Oyster Creek.

The procedures employed in the reload design and analysis are essentially the same as those described in the previously approved NEDE-24011-P and are acceptable. The procedures used to establish operating limits are similar to those previously approved and are acceptable.

2.2.3 Thermal-Hydraulic Design

The thermal-hydraulic design methods employed for Oyster Creek reloads are described in Sections 4 and 5 of NEDO-24195. These sections are essentially the same as the corresponding sections of NEDE-24011-P, which were previously reviewed and approved by the NRC for reference in the safety analysis of boiling water reactors. Since the procedures employed in the reload design and analysis are essentially the same, the staff finds the Oyster Creek application acceptable. Similarly, the procedures used to establish operating limits are similar to those previously approved and are acceptable.

2.2.4 Reactor Limits Determination

The a alyses for the rod drop accident, the fuel misloading event, and the control rod misoperation events have been reviewed as part of the Systematic Evaluation Program. These analyses have been approved [Approval letters from Crutchfield (NRC) to Finfrock (Oyster Creek) dated March 31 and April 9, 1981]. It was concluded that the analyses of these events meet present day requirements and criteria and are acceptable.

2.3 LOCA Analysis

The LOCA analysis for the Oyster Creek reactor is based upon approved codes with two minor differences. The first difference is that the approved codes were written for jet pump configurations while Oyster Creek is a non-jet pump plant. Hence, the codes had to be modified to reflect this difference in plant design. The staff concludes that the modifications made are acceptable. The second difference is that the approved codes allow for only two recirculation loops while the Oyster Creek plant has five loops. In the Oyster Creek analyses the intact loops were combined into a single loop and the broken loop was modeled as the second loop. The combining of loops in computer simulations for accidents analyses is standard industry practice and is acceptable.

It was determined that data existed on the adequacy of the distribution of the low pressure core spray. The core spray is necessary to the recovery from a LOCA and there is, in 10 CFR 50, Appendix K, an allowable value for the convective heat transfer coefficient. The experimental data showed that the spray distribution was adequate up to 55 psia of steam. However, there were no data for greater pressures and credit is taken for the initiation of core spray at 125 psia.

The licensee has provided information to show that, for pressures in excess of 40 psia, the heat transfer due to steam cooling (i.e., steam, produced by flashing and boiling, rising through the fuel bundles) is sufficient to satisfy the Appendix K requirements. In addition, it is shown that core spray could be delayed to an initiation setpoint of 40 psia for breaks up to one square foot in area without exceeding a peak clad temperature of 2200 degrees Fahrenheit. (It should be noted that the large break LOCA produces a very rapid depressurization and therefore uncertainty in spray distribution above 55 psia is of no consequence.) This was shown by comparing the heat transfer coefficient from the fuel rods to the steam which was based on the Dittus-Boelter correlation. These coefficients were referenced downward to the saturation temperature so as to be able to apply them in a previously approved code. The steam flow rate was calculated by determining the amount of boiling in the active core and adding that to the amount of steam produced by flashing that is directed to the active rods (conservatively shown to be 50 percent of the total flashing). The approved codes were then exercised. The steam cooling heat transfer coefficient remains above the spray coefficient of 1.5 for the period of concern; therefore use of the spray coefficient during that period is acceptable.

In summary, the LOCA analysis for Oyster Creek is acceptable. The computer codes were modified in an acceptable manner in order to make them applicable to a non-jet pump plant and adequate core cooling will exist to bring the plant to a safe shutdown.

2.4 Evaluation Procedure

The review of topical report NEDO-24195 has been conducted within the guidelines provided for analytical methods in the Standard Review Plan (NUREG-0800, Section 4.3). Sufficient information is provided in this report and the referenced topical reports to permit the conclusion that the methods and procedures described are state-of-the-art and are acceptable.

2.5 Regulatory Position

Based on the review of topical report NEDO-24195 described above the staff concludes that the report is suitable for reference by Oyster Creek in reload reports and other licensing actions to which it is applicable.

2.6 Technical Specification Changes and Reload 9 (Cycle 10)

Changes Due to Methodology

The staff has reviewed the proposed changes to the Technical Specifications which are intended to make them consistent with the new methodology.

Many of the changes are editorial in nature - e.g., replacing the references to methods with NEDO-24195. These are acceptable. Others include:

1. Replacing the peaking factor ratio with the ratio of the fraction of rated power to the maximum fraction of linear power density in the APRM scram and rod block functions in Section 2.3 and the bases.

2. The acceptance criterion for the rod drop accident has been changed to make it consistent with the use of the Banked Position Withdrawal Sequence and the GE analysis method.

3. The requirement for a control rod density of 3.5 percent has been deleted since new analyses are performed with the end-of-cycle, all-rods-out scram curve.

The staff finds the changes to be acceptable.

2.7 Cycle 10 Changes

The Cycle 10 core consists of 172 fresh GE P8x8R fuel assemblies and 388 partly burned Exxon Type VB assemblies. The supplemental reload submittal contains a core loading diagram.

2.7.1 Fuel Mechanical Design

The mechanical performance of the General Electric fuel in the Oyster Creek Cycle 10 core has been analyzed with the methods described in General Electric Reload Fuel Application for Oyster Creek (NEDO-24195). Our review of the GE Application for Oyster Creek is described in Section 2.1.2 of this report. The mechanical performance of the remaining Exxon fuel in the Cycle 10 core has been analyzed as part of the previous reload applications from this plant. Where the Exxon fuel affects the safety analysis of Cycle 10 specifically (this is largely due to the thermal and hydraulic characteristics of the fuel), the analyses have been performed by General Electric using Exxon fuel characteristics provided in the Oyster Creek FDSAR (and Appendix B of NEDO-24195). With regard to the fuel thermal, mechanical and materials design, the staff finds this application acceptable.

2.7.2 Nuclear Design

The nuclear design and analysis of the core was performed with the methods described in NEDO-24195. The staff has reviewed the results of the analyses. The following comments are relevant.

- The effective multiplication factor of the core is less than 0.99 at cold (20°C) xenon-free condition with the strongest control rod out of the core.

- o The standby liquid control system is capable of producing a shutdown margin of 0.044 in the cold xenon-free state.
- o The reactivity coefficients are within the range of those usually seen for BWRs.

Because the analysis was performed with acceptable methods and the results are acceptable, the staff concludes that the nuclear design and analysis of Cycle 10 is acceptable.

2.7.3 Thermal-Hydraulic Design

The thermal-hydraulic performance of the General Electric fuel in the Oyster Creek Cycle 10 core has been analyzed with the methods described in NEDO-24195 for which the licensee's application for Oyster Creek is described in Sections 2.1.4 and 2.1.5 of this report. The thermal-hydraulic performance of the remaining Exxon fuel in the Cycle 10 core has been analyzed as part of the previous reload applications from this plant. General Electric has developed a thermal-hydraulic model for non-GE fuel (Exxon 8x8 Type VB) for use in transient analyses with a mixed core of GE and Exxon fuel. The thermal-hydraulic model for the Exxon fuel design is based on the geometry of the Exxon fuel and pressure drop data for the Exxon fuel. Other thermal-hydraulic characteristics of the Exxon fuel were assumed to be identical to GE fuel. In Appendix B of NEDO-24195 the geometry of the Exxon fuel is tabulated in Table B-1 and the thermal-hydraulic model assumptions in Table B-2. The transient results for Exxon fuel are in Appendix A of NEDO-24195 along with those for the GE fuel. The staff finds the thermal-hydraulic design for this application to be acceptable.

2.7.4 Transient and Accident Analysis

The Rod Withdrawal Event - The rod withdrawal error analysis was performed with the methods described in NEDO-24195. If it is assumed that the limiting failures have occurred in the APRM rod block circuitry the change in CPR during the event is 0.27 for the GE fuel and 0.33 for the Exxon fuel. This is the limiting event for Cycle 10 and dictates the MCPR operating limit of 1.40. Analyses have been performed for less than limiting failures, for which the change in CPR is smaller, decreasing to 0.10 for no failures. However, no credit is taken for this fact in establishing cycle MCPR operating limits.

Fuel Misloading Event - The effect of the misorientation of a fuel assembly in the core has been analyzed by the methods in NEDO-24195. The limiting case results in a change in CPR of 0.20. Thus, this event is not limiting for Cycle 10.

Rod Drop Accident - The consequences of the postulated rod drop event for Cycle 10 have been calculated by methods described in NEDO-24195. The Banked Position Rod Withdrawal Sequence will be employed by the licensee. This sequence has been shown to preclude rod worths sufficient to exceed the fuel enthalpy limit of 280 calories per gram for the event. In the event that inoperable rods make adherence to the programmed sequence impossible, a rod worth check will be made to show that the maximum rod worth is less than 1.0 percent Δk. This value of rod worth has been shown to meet the 280 calories per gram criterion. This is an acceptable procedure and the staff concludes that an adequate analysis of the rod drop event has been performed.

2.7.5 Technical Specifications

Supplemental information submitted on October 28, 1983 proposed to incorporate a new scram setting for recirculation flow at 117 percent rated flow. This change is conservative since it results in a new Technical Specification (TS) requirement which did not previously exist in the current TS and is also conservative relative to the previous setting of 120 percent which had been previously established as an administrative control contained in the plant procedures. The change in MCPR limit from 1.3 to 1.4 is in the conservative direction and furnishes the maximum allowable average planar LHGR curves for 5 and 4 loop operation. Also, the change revises the control rod withdrawal sequences and establishes the maximum in sequence rod worth to be 1.0 percent Δk. This change incorporates the use of banked position withdrawal sequences which is more conservative than the previous withdrawal sequence. The staff has reviewed the TS changes proposed for Cycle 10 and conclude that they are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in 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 occupation 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 or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The staff concludes that the Oyster Creek reactor may be operated for Cycle 10 without undue risk to the health and safety of the public. This conclusion is based on the fact that acceptable methods and procedures were used to perform the design and analysis of the cycle and that the Technical Specifications have been correctly based on the results of that analysis.

The staff has also 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 or to the health and safety of the public.

5.0 ACKNOWLEDGEMENT

This evaluation was prepared by W. Brooks, J. Voglewede, and H. Balukjian.

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