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Docket No. 50-219

Jersey Central Power & Light Company
ATTN: Mr. I. R. Finfrock, Jr.
Vice President - Generation
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960

Gentlemen:

The Commission has issued the enclosed Amendment No. 23 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station. This amendment consists of changes to the Technical Specifications and is in response to your application dated November 19, 1976.

The amendment consists of changes in the Technical Specifications that will allow unloading and reloading of the core fuel during the spring outage scheduled to begin April 9, 1977. Removal of all the fuel assemblies from the Oyster Creek reactor vessel will reduce personnel radiation exposure during the planned inspection of the reactor vessel feedwater nozzles and replacement of the feedwater sparger.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Original signed by

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 23 to License DPR-16
2. Safety Evaluation
3. Federal Register Notice

cc w/encl: See page 2

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

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 23
License No. DPR-16

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

   A. The application for amendment by Jersey Central Power and Light Company (the licensee) dated November 19, 1976, 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 3.B. of Provisional Operating License No. DPR-16 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 23, are hereby incorporated in the license. The licensee 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

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 31, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 23

TO THE TECHNICAL SPECIFICATIONS

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Add pages 3.9-3 and 4.9-2.

Remove	Insert
3.2-2	3.2-2
3.2-3	3.2-3
3.9-1	3.9-1
3.9-2	3.9-2
	3.9-3
4.9-1	4.9-1
	4.9-2

Any four rod group may contain a control rod which is valved out of service provided the above requirements and Specification 3.2.A are met. Time zero shall be taken as the de-energization of the pilot scram valve solenoids.

4. Control rods which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to a shutdown condition within 48 hours unless investigation demonstrates that the cause of the failure is not due to a failed control rod drive mechanism collet housing. Inoperable control rods shall be valved out of service, in such positions that Specification 3.2.A is met. In no case shall the number of rods valved out of service be greater than six during the power operation. If this specification is not met, the reactor shall be placed in the shutdown condition.

5. Control rods shall not be withdrawn for approach to criticality unless at least three source range channels have an observed count rate equal to or greater than 3 counts per second.

6. The control rod density shall be greater than 3.5 percent during power operation.

C. Standby Liquid Control System

1. The standby liquid control system shall be operable at all times when the reactor is not shutdown by the control rods such that Specification 3.2.A is met and except as provided in Specification 3.2.C.3.

2. The standby liquid control solution shall be maintained within the volume-concentration requirement area in Figure 3.2-1 and at a temperature not less than the temperature presented in Figure 3.2-2 at all times when the standby liquid control system is required to be operable.

3. If one standby liquid control system pumping circuit becomes inoperable during the run mode and Specification 3.2.A is met the reactor may remain in operation for a period not to exceed 7 days, provided the pump in the other circuit is demonstrated daily to be operable.

D. Reactivity Anomalies

The difference between an observed and predicted control rod inventory shall not exceed the equivalent of one percent in reactivity. If this limit is exceeded and the discrepancy cannot be explained, the reactor shall be brought to the cold shutdown condition by normal orderly shutdown procedure. Operation shall not be permitted until the cause has been evaluated and appropriate corrective action has been completed. The NRC shall be notified within 24 hours of this situation in accordance with Specification 6.6.B.

BASES:

Limiting conditions of operation on core reactivity and the reactivity control systems are required to assure that the excess reactivity of the reactor core is controlled at all times. The conditions specified herein assure the capability to provide reactor shutdown from steady state and transient conditions and assure the capability of limiting reactivity insertion rates under accident conditions to values which do not jeopardize the reactor coolant system integrity or operability of required safety features.

The core reactivity limitation is required to assure the reactor can be shut down at any time when fuel is in the core. It is a restriction that must be incorporated into the design of the core fuel; it must be applied to the conditions resulting from core alterations; and it must be applied in determining the required operability of the core reactivity control devices. The basic criterion is that the core at any point in its operation be capable of being made subcritical in the cold, xenon-free condition with the operable control rod of highest worth fully withdrawn and all other operable rods fully inserted. At most times in core life more than one control rod drive could fail mechanically and this criterion would still be met.

In order to assure that the basic criterion will be satisfied an additional design margin was adopted; that the k_{eff} be less than 0.99 in the cold xenon-free condition with the rod of highest worth fully withdrawn and all others fully inserted. Thus the design requirement is $k_{eff} < 0.99$, whereas the minimum condition for operation is $k_{eff} \leq 1.0$ with the operable rod of highest worth fully withdrawn (1). This limit allows control rod testing at any time in core life and assures that the plant can be shut down by control rods alone.

The first cycle core contains boron as a burnable neutron absorber in the temporary control curtains which results in a core reactivity characteristic which increases with exposure, goes through a maximum and then decreases (2). Thus it is possible that a core could be more reactive later in the cycle than at the beginning. Satisifaction of the above criterion can be demonstrated conveniently only at the time of refuleing since it requires the core to be cold and xenon-free. The demonstration is designed to be done at these times and is such that if it is successful, the criterion is satisfied for the entire subsequent fuel cycle.

The criterion will be satisfied by demonstrating Specification 4.2.A at the beginning of each fuel cycle with the core in the cold, xenon-free condition. This demonstration will include consideration for the calculated reactivity characteristic during the following operating cycle and the uncertainty in this calculation.

The control rod drive housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. (3). The amount of reactivity

3.9 REFUELING

Applicability: Applies to fuel handling operations during refueling.

Objective: To assure that criticality does not occur during refueling.

Specification:

A. Fuel shall not be loaded into a reactor core cell unless the control rod in that core cell is fully inserted.

B. During core alterations the reactor mode switch shall be locked in the REFUEL position.

C. The refueling interlocks shall be operable with the fuel grapple hoist loaded switch set at ≤485 lb. during the fuel handling operations with the head off the reactor vessel. If the frame-mounted auxiliary hoist, the trolley-mounted auxiliary hoist or the service platform hoist is to be used for handling fuel with the head off the reactor vessel the load limit switch on the hoist to be used shall be set at ≤400lb.

D. During core alterations the source range monitor nearest the alteration shall be operable.

E. Removal of one control rod or rod drive mechanism may be performed provided that all the following specifications are satisfied.

1. The reactor mode switch is locked in the refuel position.

2. At least two (2) source range monitor (SRM) channels shall be operable and inserted to the normal operation level. One of the operable SRM channel detectors shall be located in the core quadrant where the control rod is being removed and one shall be located in an adjacent quadrant.

F. Removal of any number of control rods or rod drive mechanisms may be performed provided that all the following specifications are satisfied:

1. The reactor mode switch is locked in the refuel position and all refueling interlocks are operable as required in Specification 3.9.C. The refueling interlocks associated with the control rods being withdrawn may be bypassed as required after the fuel assemblies have been removed from the core cell surrounding the control rods as specified in 4, below.

void after cycle 6 to 7 core offload- reload

2. At least two (2) source range monitor (SRM) channels shall be operable and inserted to the normal operation level. One of the operable SRM channel detectors shall be located in the core quadrant where a control rod is being removed and one shall be located in an adjacent quadrant.

3. All other control rods are fully inserted with the exception of one rod which may be partially withdrawn not more than two notches to perform refueling interlock surveillance.

4. The four fuel assemblies are removed from the core cell surrounding each control rod or rod drive mechanism to be removed.

5. The core is subcritical by at least 0.25% Δk, plus equivalent reactivity for the effect of any B_4C settling in inverted tubes present in the core, with the most reactive remaining control rod withdrawn.

void after cycle 6 - 7 core offload-reload

G. With any of the above requirements not met, cease core alterations or control rod removal as appropriate, and initiate action to satisfy the above requirements.

Basis:

During refueling operations, the reactivity potential of the core is being altered. It is necessary to require certain interlocks and restrict certain refueling procedures such that there is assurance that inadvertent criticality does not occur.

Addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks (1) on rod withdrawal and movement of the refueling platform. When the mode switch is in the "Refuel" position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. Likewise, if the refueling platform is over the core with fuel on a hoist control rod motion is blocked by the interlocks. With the mode switch in the refuel position only one control rod can be withdrawn (1,2).

The one rod withdrawal interlock may be bypassed in order to allow multiple control rod removal for repair, modifications, or core unloading. The requirements for simultaneous removal of more than one control rod are more strigent than the requirements for removal of a single control rod, since in the latter case Specification 3.2.A assures that the core will remain subcritical.

void after cycle 6-7 core offload-reload

Fuel handling is normally conducted with the fuel grapple hoist. The total load on this hoist when the interlock is required consists of the weight of the fuel grapple and the fuel assembly. This total is approximately 773 lbs. in the extended position in comparison to the load limit of 485 lbs. Provisions have also been made to allow fuel handling with either of the three auxiliary hoists and still maintain the refueling interlocks. The 400 lb load trip setting on these hoists is adequate to trip the interlock when one of the more than 600 lb. fuel bundles is being handled.

void after 1977
core offload-reload

The source range monitors provide neutron flux monitoring capabilities when the reactor is in the refueling and shutdown modes (3). Specification 3.9.D assures that the neutron flux is monitored as close as possible to the location where fuel or controls are being moved. Specifications 3.9.E and F require the operability of at least two source range monitors when control rods are to be removed.

References:

(1) FDSAR, Volume I, Section VII-7.2.5
(2) FDSAR, Volume I, Section XIII-2.2
(3) FDSAR, Volume I, Sections VII-4.2.2 and VII-4.3.1

4.9 REFUELING

Applicability: Applies to the periodic testing of those interlocks and instruments used during refueling.

Objective: To verify the operability of instrumentation and interlocks in use during refueling.

Specification:

A. The refueling interlocks shall be tested prior to any fuel handling with the head off the reactor vessel, at weekly intervals thereafter until no longer required and following any repair work associated with the interlocks.

B. Prior to beginning any core alterations, the source range monitors (SRMs) shall be calibrated. Thereafter, the SRM's will be checked daily, tested monthly and calibrated every 3 months until no longer required.

C. Within four (4) hours prior to the start of control rod removal pursuant to Specification 3.9.E verify:

1. That the reactor mode switch is locked in the refuel position and that the one rod out refueling interlock is operable.

2. That two (2) SRM channels, one in the core quadrant where the control rod is being removed and one in an adjacent quadrant, are operable and inserted to the normal operation level.

D. Verify within four (4) hours prior to the start of control rod removal pursuant to Specification 3.9.F and at least once per 24 hours thereafter, until replacement of all control rods or rod drive mechanisms and all control rods are fully inserted that:

1. the reactor mode switch is locked in the refuel position and the one rod out refueling interlock is operable.

2. Two (2) SRM channels, one in the core quadrant where a control rod is being removed and one in an adjacent quadrant, are operable and fully inserted.

3. All control rods not removed are fully inserted with the exception of one rod which may be partially withdrawn not more than two notches to perform refueling interlock surveillance.

4. The four fuel assemblies surrounding each control rod or rod drive mechanism being removed or maintained at the same time are removed from the core cell.

void after cycle 6 - 7 offload-reload

void after cycle 6 - 7 offload-reload

E. Verify prior to the start of removal of control rods pursuant to Specification 3.9.F that Specification 3.9.F.5 will be met.

F. Following replacement of a control rod or rod drive mechanism removed in accordance with Specification 3.9.F, prior to inserting fuel in the control cell, verify that the bypassed refueling interlocks associated with that rod have been restored and that the control rod is fully inserted.

Basis:

The refueling interlocks [1] are required only when fuel is being handled and the head is off the reactor vessel. A test of these interlocks prior to the time when they are needed is sufficient to ensure that the interlocks are operable. The testing frequency for the refueling interlocks is based upon engineering judgment and the fact that the refueling interlocks are a backup for refueling procedures.

The SRM's [2] provide neutron monitoring capability during core alterations. A calibration using external testing equipment to calibrate the signal conditioning equipment prior to use is sufficient to ensure operability. The frequencies of testing, using internally generated test signals, and recalibration, if the SRM's are required for an extended period of time, are in agreement with other instruments of this type which are presented in Specification 4.1.

The surveillance requirements for control rod removal assure that the requirements of Specification 3.9 are met prior to initiating control rod removal and at appropriate intervals thereafter.

References: (1) FDSAR, Volume I, Section VII-7-2.5
(2) FDSAR, Volume I, Sections VII-4.2.2 and VII-4-5.1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 23 TO LICENSE NO. DPR-16

JERSEY CENTRAL POWER AND LIGHT COMPANY

OYSTER CREEK NUCLEAR POWER PLANT UNIT NO. 1

DOCKET NO. 50-219

Introduction

By letter dated November 19, 1976, Jersey Central Power & Light Company requested changes to the Technical Specifications of Provisional Operating License No. DPR-16. At a meeting in Bethesda, Maryland, on January 12, 1977, of NRC & JCP&L representatives, the proposed changes and supportive criticality calculations were discussed. The proposed changes have been reviewed by the Oyster Creek on site (Plant Operating Review Committee) and offsite (General Office Review Committee) safety review committees.

The changes to the Technical Specifications will allow unloading and reloading of the core fuel during the spring outage scheduled to begin on April 9, 1977. Transfer of all the fuel assemblies in the Oyster Creek reactor vessel to the spent fuel pool storage facility will reduce personnel radiation exposure during the planned inspection of the reactor vessel feedwater nozzles and replacement of the feedwater sparger. We have evaluated core criticality considerations during fuel movement and each of the changes to the Technical Specifications that have been proposed by JCP&L.

Evaluation

Core Criticality

Defueling and subsequent refueling leads to unusual core configurations. The reasons for this are (1) relatively few blade guides (used to provide lateral support to the control blade in a defueled cell) are available and (2) the licensee desires to use the installed startup range monitors (SRM's), rather than dunking chambers, i.e., waterproof core chambers temporarily inserted into the reactor vessel, to monitor the core during alterations. The SRMs must be within the configuration of fuel assemblies remaining in the core to be effective.

The order of fuel assembly removal that results leads to configurations with moderator-filled cavities (cells from which both fuel and the control blade have been removed) imbedded in the core. The increased moderation in a defueled cell alters the worths of that cell's control blade and also the neighboring control blades. The question of safety significance for such configurations is: will the negative reactivity introduced by removing the four fuel assemblies be greater than the positive reactivity introduced by removing the associated control blade? The present technical specifications (Specification 3.2.D) require a reactivity shutdown design margin so that the core is at least 1.00% subcritical with the highest worth control blade withdrawn and all other control blades fully inserted. In this evaluation the highest worth control blade is assumed to be withdrawn in addition to the control blade removed from the adjacent defueled cell.

To evaluate the effect on shutdown margin, the licensee has performed PDQ computer code calculations for various configurations. Each configuration was calculated for a "nominal" case, where all assemblies were at 10,000 MWD/t burnup, and for a "conservative" case, where the exposures of the assemblies surrounding the high worth rod were reduced to increase the reactivity worth of the rod. The four configurations studied were:

Fully loaded core,	All rods in.
Fully loaded core,	Hot rod out.
Adjacent cell defueled,	All rods in, except rod in defueled cell.
Adjacent cell defueled,	Hot rod out, rod in defueled cell out.

The net results indicate an increase of 1.78% in the shutdown margin for the conservative case (2.78% shutdown margin compared to 1% for the as designed core) when the cell adjacent to the highest worth control blade is defueled.

Rod worths calculated in the "nominal" case agree well with actual rod worths observed in the Oyster Creek reactor. Added assurance is thereby provided that the calculations are conservative and therefore acceptable.

Our evaluation considers the applicability of the proposed Technical Specification only for the defueling and refueling operation to be conducted at the end of core cycle 6. Calculation of an actual criticality for the Oyster Creek reactor must be provided by JCP&L prior to any defueling operation subsequent to the one currently scheduled for April 1977. This further analysis is necessary to verify the calculational techniques at a different burnup, and thus provide a basis for use of these methods described by JCP&L for further off-loading and reloading in the future.

The NRC staff agrees that the "conservative" case studied bounds the configurations produced by the proposed order of defueling and refueling for this cycle of the Oyster Creek reactor. The staff further finds that the calculational tools used for this study are adequate for the task. Therefore, for the cycle 6 to 7 order of defueling & refueling proposed by the licensee, the staff finds the proposed Technical Specification change relating to criticality acceptable.

Control Rod Withdrawal Interlocks

Refueling interlocks are provided as procedural backup to prevent the addition of reactivity to the core that could result in unplanned criticality. When in the REFUEL mode, refueling interlocks, in addition to other functions, prevent withdrawal of more than one control rod and under certain conditions prevent withdrawal (removal) of any control rods. We have concluded in the preceding section, based on PDQ calculated results, that when the four fuel assemblies in core positions adjacent to a control rod are removed, the reactivity withdrawn is greater than the reactivity inserted when the control rod associated with the four fuel assemblies is withdrawn. In other words the shutdown reactivity margin is greater, and the core is less reactive. Therefore, we have also concluded that the proposed Technical Specification changes to allow bypassing of refueling interlocks for a selected control rod after the four adjacent fuel assemblies have been withdrawn are acceptable.

Control Rod Interlock Bypass Error

If the interlock on a control rod is unintentionally bypassed (i.e., the wrong control rod interlock is removed after the fuel and control rod have been withdrawn from a cell), the mistake will be evident as soon as an attempt is made to remove another fuel assembly or control rod from the core. Refueling interlocks will block such action until the mistake is corrected. On this basis we have concluded that the proposed changes to the Technical Specifications are acceptable.

Refueling Accident

According to the FDSAR the reactor core is designed so that it remains subcritical with one of the control rods fully withdrawn even if it is assumed that a fuel assembly is dropped into an empty fuel space in an otherwise fully constituted core. The control rod withdrawal interlock system reinforces administrative procedures to assure that such an unplanned criticality is never achieved. We have concluded that the proposed Technical Specification changes to allow core defueling and reloading do not introduce the potential for accidents that have not been previously evaluated and approved. On this basis the potential for unplanned core criticality during the unloading and reloading of fuel assemblies is not changed significantly and the proposed Technical Specification changes are therefore acceptable.

The potential for unplanned criticality in the spent fuel pool has been reexamined because of the planned increase in fuel pool storage capacity (refer to Amendment No. 22 dated March 30, 1977) and found to be acceptably low because the neutron multiplication factor, K_{eff}, is less than the NRC acceptance criteria of 0.95.

We have therefore concluded that the proposed Technical Specification changes related to unloading and reloading the core considering storage of the off-loaded fuel in the spent fuel pool are acceptable.

End of Cycle Control Rod Density

The requirement of 3.5% control rod density (Technical Specification 3.2.B.6) is based on the control rod scram reactivity response used to evaluate abnormal operating transients. It is not applicable in the shutdown or refuel mode. The proposed Technical Specification is consistent with the original intent and is therefore acceptable.

Environmental Considerations

We have determined that this amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Dated: March 31, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-219

JERSEY CENTRAL POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT

TO PROVISIONAL OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 23 to Provisional Operating License No. DPR-16 issued to Jersey Central Power & Light Company which revised Technical Specifications for operation of the Oyster Creek Nuclear Generating Station, located in Ocean County, New Jersey. The amendment is effective as of its date of issuance.

The amendment consists of changes in the Technical Specifications that will allow unloading and reloading of the core fuel during the spring outage scheduled to begin April 9, 1977. Removal of all the fuel assemblies from the Oyster Creek reactor vessel will reduce personnel radiation exposure during the planned inspection of the reactor vessel feedwater nozzles and replacement of the feedwater sparger.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated November 19, 1976, (2) Amendment No. 23 to License No. DPR-16, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Ocean County Library, Brick Township Branch, 401 Chambers Bridge Road, Brick Town, New Jersey. 08723. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 31 day of March 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors