FEB 4 1977

Distribution
Docket
ORB #3
NRC PDR
Local PDR
VStello
KRGoller/TJCarter
CParrish
GLear
JShea
DVerrelli
Attorney, OELD
OI&E (5)
BJones (4)
BScharf (10)
JMcGough
DEisenhut
ACRS (16)
TBAbernathy
JRBuchanan
D Ross

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. 21 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 January 11, 1977.

The amendment consists of changes in the Technical Specifications that will allow alternate emergency core cooling provisions with fuel in the vessel and permit draining the water from the torus for specified conditions.

Copies of the related Safety Evaluation and the FEDERAL REGISTER Notice also are enclosed.

Sincerely,

Original signed by J Shea for

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

Enclosures:
1. Amendment No. 21 to License DPR-16
2. Safety Evaluation
3. FEDERAL REGISTER NOTICE

cc w/encls:
See next page

*SEE PREVIOUS YELLOW FOR CONCURRENCES

OFFICE	ORB #3	ORB #3	OELD	ORB #3		
SURNAME	*CParrish	Shea/Verrelli	[illegible]	GLear		
DATE	1/26/77	2/2/77	2/2/77	2/4/77		

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The Commission has issued the enclosed Amendment No. 21 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 January 11, 1977.

The amendment consists of changes in the Technical Specifications that will permit draining the water from the torus whenever all of the fuel assemblies are removed from the reactor or allow alternate emergency core cooling provisions with fuel in the vessel as long as the coolant system is depressurized.

Copies of the related Safety Evaluation and the FEDERAL REGISTER Notice also are enclosed.

Sincerely,

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

Enclosures:
1. Amendment No. 21 to License DPR-16
2. Safety Evaluation
3. FEDERAL REGISTER Notice

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See next page

OFFICE	ORB #3	ORB #3	OELD	ORB #3		
SURNAME	CParrish CP	Shea/Verrelli		GLear		
DATE	1/26/77	1/ /77				

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166 cc:

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Brick Township Branch
401 Chambers Bridge Road
Brick Town, New Jersey 08723

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. 21
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 January 11, 1977, 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 a change to the Technical Specifications as indicated in the attachment to this license amendment.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James J Shea

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 4, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 21

TO THE TECHNICAL SPECIFICATIONS

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace pages 3.4-1, 3.4-1a, 3.4-5, 3.5-1, 3.5-2, 3.5-3 and 3.5-4a with the attached revised pages. Add pages 3.4-1b, 3.4-6 and 3.5-1a

3.4 EMERGENCY COOLING

Applicability: Applies to the operating status of the emergency cooling systems.

Objective: To assure operability of the emergency cooling systems.

Specification: A. Core Spray System

1. The core spray system shall be operable at all times with **irradiated** fuel in the reactor vessel, except as otherwise specified in this section.

2. The absorption chamber water volume shall be at least 82,000 ft.^{3} in order for the core spray system to be considered operable.

3. If one core spray system loop or its core spray header ΔP instrumentation becomes inoperable during the run mode, the reactor may remain in operation for a period not to exceed 7 days provided the remaining loop has no inoperable components and is demonstrated daily to be operable.

4. If one of the redundant active loop components in the core spray system becomes inoperable during the run mode, the reactor may remain in operation for a period not to exceed 15 days provided the other similar component in the loop is demonstrated daily to be operable. If two of the redundant active loop components become inoperable, the limits of Specification 3.4.A shall apply.

5. During the period when one diesel is inoperable, the core spray equipment connected to the operable diesel shall be operable.

6. If Specifications 3.4.A.3, 3.4.A.4, and 3.4.A.5 are not met, the reactor shall be placed in the cold shutdown condition. If the core spray system becomes inoperable, the reactor shall be placed in the cold shutdown condition and no work shall be performed on the reactor or its connected systems which could result in lowering the reactor water level to less than 4'8" above the top of the active fuel.

7. If necessary to accomplish maintenance or modifications to the core spray systems, their power supplies or water supplies, reduced system availability is permitted when the reactor is: (a) maintained in the cold shutdown condition or (b) in the refuel mode with the reactor coolant system maintained at less than 212°F and vented, and (c) no work is performed on the reactor vessel and connected systems that could result in lowering the reactor water level to less than 4'8" above the top of the active fuel. Reduced Core Spray System Availability is minimally defined as follows:

a. At least one core spray pump, and system components necessary to deliver rated core spray to the reactor vessel, must remain operable to the extent that the pump and any necessary valves can be started or operated from the control room or from local control stations.

b. The fire protection system is operable, and

c. These systems are demonstrated to be operable on a weekly basis.

8. If necessary to accomplish maintenance or modifications to the core spray systems, their power supplies or water supplies, reduced system availability is permitted when the reactor is in the refuel mode with the reactor coolant system maintained at less than 212°F or in the startup mode for purposes of low power physics testing. Reduced core spray system availability is defined as follows:

a. At least one core spray pump in each loop, and system components necessary to deliver rated core spray to the reactor vessel, must remain operable to the extent that the pump and any necessary valves in each loop can be started or operated from the control room or from local control stations.

b. The fire protection system is operable and,

c. Each core spray pump and all components in 3.4.A.8a are demonstrated to be operable every 72 hours.

9. If Specifications 3.4.A.7 and 3.4.A.8 cannot be met, the requirements of Specifications 3.4.A.6 will be met and work will be initiated to meet minimum operability requirements of 3.4.A.7 and 3.4.A.8.

10. The core spray system is not required to be operable when the following conditions are met:

a. The reactor mode switch is locked in the refuel or shutdown position.

b.(1) There is an operable flow path capable of taking suction from the condensate storage tank and transferring water to the reactor vessel, and

(2) The fire protection system is operable.

c. The reactor vessel head, fuel pool gate, and separator-dryer pool gates are removed and the water level is above elevation 117 feet.

d.(1) Operations which have the potential for draining the reactor vessel below 4'8" above the top of the active fuel are prohibited, or

(2) At least one core spray pump, and system components necessary to deliver rated core spray flow to the reactor vessel, must remain operable to the extent

that the pump and any necessary valves can be started or operated from the control room or from local control stations and the torus is mechanically intact.

B. Automatic Depressurization System

1. Five electromatic relief valves of the automatic depressurization system shall be operable when the reactor is pressurized above 110 psig, except as specified in 3.4.B.2.

spray pump capable of full rated flow and the 72 hour operability demonstration of both core spray pumps is specified.

Specification 3.4.A.10 allows the core spray system to be inoperable in the cold shutdown or refuel modes if the reactor cavity is flooded and the spent fuel pool gates are removed and a source of water supply to the reactor vessel is available. Water would then be available to keep the core flooded.

The relief valves of the automatic depressurization system enable the core spray system to provide protection against the small break in the event the feedwater system is not active.

The containment spray system is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. The flow from one pump in either loop is more than ample to provide the required heat removal capability (2). The emergency service water system provides cooling to the containment spray heat exchangers and, therefore, is required to provide the ultimate heat sink for the energy release in the event of a loss-of-coolant accident. The emergency service water pumping requirements are those which correspond to containment cooling heat exchanger performance implicit in the containment cooling description. Since the loss-of-coolant accident while in the cold shutdown condition would not require containment spray, the system may be deactivated to permit integrated leak rate testing of the primary containment while the reactor is in the cold shutdown condition.

The control rod drive hydraulic system can provide high pressure coolant injection capability. For break sizes up to 0.002 ft^2, a single control rod drive pump with flow of 110 gpm is adequate for maintaining the water level nearly five feet above the core, thus alleviating the necessity for auto-relief actuation (3).

The core spray main pump compartments and containment spray pump compartments were provided with water-tight doors.(4) Specification 3.4.E ensures that the doors are in place to perform their intended function.

Similarly, since a loss-of-coolant accident when primary containment integrity is not being maintained would not result in pressure build-up in the drywell or torus, the system may be made inoperable under these conditions. This prevents possible personnel injury associated with contact with chromated torus water.

References

(1) Licensing Application, Amendment
(2) Licensing Application, Amendment 32, Question 3
(3) Licensing Application, Amendment 18, Question 1
(4) Licensing Application, Amendment 18, Question 4

3.5 CONTAINMENT

Applicability: Applies to the operating status of the primary and secondary containment systems.

Objective: To assure the integrity of the primary and secondary containment system.

Specification: A. Primary Containment

1. At any time that the nuclear system is pressurized above atmospheric or work is being done which has the potential to drain the vessel and irradiated fuel is in the vessel, the suppression pool water volume and temperature shall be maintained within the following limits.

   a. Maximum water volume - 92,000 ft^3
   b. Minimum water volume - 82,000 ft^3
   c. Maximum water temperature

      (1) During normal power operation - 95°F

      (2) During testing which adds heat to the suppression pool, the water temperature shall not exceed 10F° above the normal power operation limit specified in (1) above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in (1) above within 24 hours.

      (3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110°F. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in (1) above.

      (4) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 180 psig at normal cooldown rates if the pool temperature reaches 120°F.

2. Maintenance and repair, including draining of the suppression pool, may be performed provided that the following conditions are satisfied:

   a. The reactor mode switch is locked in the refuel or shutdown position.

   b. (1) There is an operable flow path capable of taking suction from the condensate storage tank and transferring water to the reactor vessel, and

      (2) The fire protection system is operable.

c. The reactor vessel head, fuel pool gate, and separator-dryer pool gates are removed and the water level is above elevation 117 feet.

d. At least one core spray pump, and system components necessary to deliver rated core spray flow to the reactor vessel, must remain operable to the extent that the pump and any necessary valves can be started or operated from the control room or from local control stations, and the torus is mechanically intact.

3. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.

4. Reactor Building to Suppression Chamber Vacuum Breaker System

   a. Except as specified in Specification 3.5.A.3.b below, two reactor building to suppression chamber vacuum breakers in each line shall be operable at all times when primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the air-operated vacuum breakers shall not exceed 0.5 psid. The vacuum breakers shall move from closed to fully open when subjected to a force equivalent of not greater than 0.5 psid acting on the vacuum breaker disc.

   b. From the time that one of the reactor building to suppression chamber vacuum breakers is made or found to be inoperable, the vacuum breaker shall be locked closed and reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is made operable sooner, provided that the procedure does not violate primary containment integrity.

   c. If the limits of Specification 3.5.A.3.a are exceeded, reactor shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

5. Pressure Suppression Chamber - Drywell Vacuum Breakers

   a. When primary containment is required, all suppression chamber - drywell vacuum breakers shall be operable except during testing and as stated in Specification 3.5.A.4.b and c, below. Suppression chamber - drywell vacuum breakers shall be considered operable if:

      (1) The valve is demonstrated to open from closed to fully open with the applied force at all valve positions not exceeding that equivalent to 0.5 psi acting on the suppression chamber face of the valve disk.

      (2) The valve disk will close by gravity to within not greater than 0.10 inch of any point on the seal surface of the disk when released after being opened by remote or manual means.

      (3) The position alarm system will annunciate in the control room if the valve is open more than 0.10 inch at any point along the seal surface of the disk.

b. Two of the fourteen suppression chamber - drywell vacuum breakers may be inoperable provided that they are secured in the closed position.

c. One position alarm circuit for each operable vacuum breaker may be inoperable for up to 15 days provided that each operable suppression chamber - drywell vacuum breaker with one defective alarm circuit is physically verified to be closed immediately and daily during this period.

6. After completion of the startup test program and demonstration of plant electrical output, the primary containment atmosphere shall be reduced to less than 5.0% O_2 with nitrogen gas within 24 hours after the reactor mode selector switch is placed in the run mode. Primary containment deinerting may commence 24 hours prior to a scheduled shutdown.

7. If specifications 3.5.A.1.a, b, c(1) and 3.5.A.2 through 3.5.A.5 cannot be met, reactor shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

8. Shock Suppressors (Snubbers)

a. During all modes of operation except cold shutdown and refuel, all safety related snubbers listed in Table 3.5.1 shall be operable except as noted 3.5.A.7.b, c and d below.

b. From and after the time that a snubber is determined to be inoperable, continued reactor operation is permissible only during the succeeding 72 hours unless the snubber is sooner made operable or replaced.

c. If the requirements of 3.5.A.7.a and 3.5.A.7.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 36 hours.

d. If a snubber is determined to be inoperable while the reactor is in the shutdown or refuel mode, the snubber shall be made operable or replaced prior to reactor startup.

e. Snubbers may be added to safety related systems without prior License Amendment to Table 3.6.1 provided that a revision to Table 3.6.1 is included with the next License Amendment request.

B. <u>Secondary Containment</u>

1. Secondary containment integrity shall be maintained at all times unless all of the following conditions are met.

a. The reactor is subcritical and Specification 3.2.A is met.

b. The reactor is in the cold shutdown condition.

c. The reactor vessel head or the drywell head are in place.

and fission products which could be released from any fuel failures resulting from the accident. If the reactor coolant is not above 212°F, there would be no pressure rise in the containment. In addition, the coolant cannot be expelled at a rate which could cause fuel failure to occur before the core spray system restores cooling to the core. Primary containment is not needed while performing low power physics tests since the rod worth minimizer would limit the worst case rod drop accident to 1.5%Δk. This amount of reactivity addition is insufficient to cause fuel damage.

The absorption chamber water volume provides the heat sink for the reactor coolant system energy released following the loss-of-coolant accident. The core spray pumps and containment spray pumps are located in the corner rooms and due to their proximity to the torus, the ambient temperature in those rooms could rise during the design basis accident. Calculations (7) made, assuming an initial torus water temperature of 100°F and a minimum water volume of 82,000 ft^3, indicate that the corner room ambient temperature would not exceed the core spray and containment spray pump motor operating temperature limits, and, therefore, would not adversely affect the long term core cooling capability. The maximum water volume limit allows for an operating range without significantly affecting accident analyses with respect to free air volume in the absorption chamber. For example, the containment capability (8) with a maximum water volume of 92,000 ft^3 is reduced by not more than 3.5% metal-water reaction below the capability with 82,000 ft^3.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

The technical specifications allow for torus repair work or inspections that might require draining of the suppression pool when all irradiated fuel is removed or when the potential for draining the reactor vessel has been minimized. This specification also provides assurance that the irradiated fuel has an adequate cooling water supply for normal and emergency conditions with the reactor mode switch in shutdown or refuel whenever the suppression pool is drained for inspection or repair.

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and suppression chamber and reactor building so that the containment external design pressure limits are not exceeded.

The vacuum relief system from the reactor building to the pressure suppression chamber consists of two 100% vacuum relief breaker subsystems (2 parallel sets of 2 valves in series). Operation of either subsystem will maintain the containment external pressure less than the external design pressure; the external design pressure of the drywell is 2 psi; the external design pressure of the suppression chamber is 1 psi (FDSAR Amendment 15, Section II).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 21 TO LICENSE NO. DPR-16

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER-CREEK NUCLEAR POWER PLANT UNIT NO. 1

DOCKET NO. 50-219

INTRODUCTION

By letter dated January 11, 1977, Jersey Central Power & Light Company requested changes to the Technical Specifications of Provisional Operating License No. DPR-16. The changes reviewed and approved by the Oyster Creek on site and off site safety review committees would permit:

1. elimination of suppression pool water level and temperature limits if there is no irradiated fuel in the reactor vessel.

2. making the core spray system inoperative if the following conditions are met:

   a) The reactor mode switch is locked in the refuel or shutdown position.

   b) 1. There is an operable flow path capable of taking suction from the condensate storage tank and transferring water to the reactor vessel, and

      2. The fire protection system is operable.

   c) The reactor vessel head, fuel pool gate, and separator-dryer pool gate are removed and the water level is above elevation 117 feet.

   d) 1. Operations which have the potential for draining the reactor vessel below 4' 8" above the top of the active fuel are prohibited, or

      2. At least one core spray pump, and system components necessary to deliver rated core spray flow to the reactor vessel, must remain operable to the extent that the pump and any necessary valves can be started or operated from the control room or from local control stations, and the torus is mechanically intact.

3. draining the torus water <u>with</u> irradiated fuel assemblies in the vessel provided:

   a) The reactor mode switch is locked in the refuel or shutdown position.

   b) 1. There is an operable flow path capable of taking suction from the condensate storage tank and transferring water to the reactor vessel, and

      2. The fire protection system is operable.

   c) The reactor vessel head, fuel pool gate, and separator-dryer pool gates are removed and the water level is above elevation 117 feet.

   d) At least one core spray pump, and system components necessary to deliver rated core spray flow to the reactor vessel, must remain operable to the extent that the pump and any necessary valves can be started or operated from the control room or from local control stations, and the torus is mechanically intact, and

4. connecting the core spray equipment to the operable emergency diesel generator when the other diesel generator is inoperable.

<u>EVALUATION</u>

The water in the torus is primarily the emergency heat sink for any postulated transient or accident condition that releases energy from the primary coolant system through relief and safety valves or coolant system breaks.

The suppression pool (torus water) receives energy during these transients or accidents in the form of steam and water from the reactor pressure relief discharge piping or from the drywell vent system downcomers following release of energy into the drywell. The steam discharges into the suppression pool several feet below the surface of the water and condenses.

If there are no irradiated fuel assemblies in the reactor vessel there is no source of heat, no basis for torus water level and temperature limits and no need for the torus water as an emergency heat sink. The water can be drained from the torus whenever all irradiated fuel assemblies have been removed from the reactor vessel without increasing the probability of reactor accidents or the risk to the health and safety of the public.

When the reactor mode switch is locked in the "refuel" or "shutdown" position with the primary system depressurized and cold (less than 212°F) the suppression pool water is no longer required as an emergency heat sink to condense steam from the pressurized coolant system. However, the torus water is also the normal source of water for the emergency core spray system. If an alternate source of water is provided for equivalent emergency core cooling under these conditions there is no increase in the probability of an accident or risk to the health and safety of the public. Thus, the emergency core spray system may be made inoperative with certain restrictions that have been specified and the water may be drained from the torus as long as the torus remains mechanically intact.

These proposed technical specification changes conform to the NRC Standard Technical Specifications for the GE Boiling Water Reactors dated August 15, 1976.

The alignment of diesel generators to the core spray systems was revised to meet the single failure criteria in accordance with Amendment No. 8, dated May 24, 1975 and JCP&L letters dated June 24, 1975 and July 15, 1975. This proposed change to the technical specifications, unrelated to the other changes to permit draining of the torus, is necessary to reflect the revised diesel generator-core spray equipment alignment that has already been completed.

ENVIRONMENTAL CONSIDERATION

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: February 4, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-219

JERSEY CENTRAL POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT

TO PROVISIONAL OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 21 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 will allow alternate emergency core cooling provisions with fuel in the vessel and permit draining the water from the torus for specified conditions.

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 January 11, 1977, (2) Amendment No. 21 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 4th day of February 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

James J. Shea

James J. Shea, Acting Chief
Operating Reactors Branch #3
Division of Operating Reactors