REGULATORY DOCKET FILE COPY

June 13, 1980

Docket No. 50-286

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Mr. George T. Berry, President
and Chief Operating Officer
Power Authority of the State of New York
10 Columbus Circle
New York, New York 10019

Dear Mr. Berry:

The Commission has issued the enclosed Amendment No. 30 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. This amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated May 6, 1980.

The amendment adds surveillance requirements for the containment vent and purge system and adds precautions for the control of heavy loads over the reactor when the vessel head is removed.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by:
S. A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 30 to DPR-64
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

~~8007100103~~

TAC 8761

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DATE	05/28/80	05/28/80	6/2/80	6/2/80	6/9/80	6/2/80

#9

June 9, 1980

Note to Lenny Olshan

INDIAN POINT 3 - SAFETY EVALUATION

On page 4 there is some confusion created by the reference to spent fuel in the last sentence of the section. This confusion arises because in a previous paragraph on the same page an analysis of a fuel handling accident in the spent fuel pool is discussed. I would suggest: 1) add to first full paragraph on page 4, first sentence "spent fuel pool, even though it is located outside containment at this facility."; and 2) make it clear in the last sentence that the spent fuel referred to is spent fuel in the reactor vessel.

There is no legal objection the letter, the amendment, and the Federal Register notice.

Janice E. Moore
Counsel for NRC Staff

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

June 13, 1980

Docket No. 50-286

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Power Authority of the State of New York
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See next page

~~8007100103~~

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New York, New York 10007

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

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-286

INDIAN POINT STATION UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 30
License No. DPR-64

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

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated May 6, 1980, 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.

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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 Facility Operating License No. DPR-64 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 30 , 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

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 13, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 30

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
iii	iii
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3.8-2	3.8-2
3.8-4	3.8-4
3.8-5	3.8-5
	4.13-1

Section	Title	Page
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3.8 Refueling, Fuel Handling and Storage

Applicability

Applies to operating limitations during refueling, fuel handling, storage operations, and when heavy loads are moved over the reactor when the head is removed.

Objective

To ensure that no incident could occur during refueling, fuel handling, and storage operations that would adversely affect public health and safety.

Specification

A. During handling operations, reactor vessel head removal or installation, or movement of heavy loads over the reactor vessel with the head removed, the following conditions shall be satisfied:

1. The equipment door and at least one door in each personnel air lock shall be properly closed. In addition, at least one isolation valve shall be operable or locked closed in each line penetrating the containment and which provides a direct path from containment atmosphere to the outside.

2. Radiation levels in the containment and spent fuel storage areas shall be monitored continuously.

3. The core subcritical neutron flux shall be continuously monitored by the two source range neutron monitors, each with continuous visual indication in the control room and one with audible indication in the containment available whenever core geometry is being changed. When core geometry is not being changed, at least one source range neutron flux monitor shall be in service.

4. At least one residual heat removal pump and heat exhanger shall be operating except during those core alterations in which the residual heat removal flow interferes with component positioning.

5. During reactor vessel head removal and while loading and unloading fuel in the reactor, T_{avg} shall be $\leq 140°$ and the minimum boron concentration sufficient to maintain the reactor subcritical by at least 10% $\Delta k/k$. The required boron concentration shall be verified by chemical analysis daily.

6. Direct communication between the control room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

7. The containment vent and purge system, including the radiation monitors which initiate isolation, shall be tested and verified to be operable within 100 hours prior to refueling operations.

8. No movement of irradiated fuel in the reactor shall be made until the reactor has been subcritical for at least 120 hours. In addition, movement of fuel in the reactor before the reactor has been subcritical for equal to or greater than 365 hours will necessitate operation of the Containment Building Vent and Purge System through the HEPA filters and charcoal adsorbers. For this case operability of the Containment Building Vent and Purge System shall be established in accordance with Section 4.13 of the Technical Specifications. In the event that more than one region of fuel (72 assemblies) is to be discharged from the reactor, those assemblies in excess of one region shall not be discharged before an interval of 400 hours has elapsed after shutdown.

9. Whenever movement of irradiated fuel is being made, the minimum water lever in the area of movement shall be maintained 23 feet over the top of irradiated fuel assemblies seated within the reactor pressure vessel.

10. Hoists or cranes utilized in handling irradiated fuel shall be dead-load tested before movement begins. The load assumed by the hoists or cranes for this test must be equal to or greater than maximum load to be assumed by the hoists or cranes during the refueling operation. A thorough visual inspection of the hoists or cranes shall be made after the dead-load test and prior to fuel handling. A test of interlocks shall also be performed.

11. The fuel storage building emergency ventilation system shall be operable whenever irradiated fuel is being handled within the fuel storage building. The emergency ventilation system may be inoperable when irradiated fuel is in the fuel storage building, provided irradiated fuel is not being handled and neither the spent fuel cask nor the cask crane are moved over the spent fuel pit during the period of inoperability.

6. The fuel storage building emergency ventilation system shall be operable whenever irradiated fuel is being handled within the fuel storage building. The emergency ventilation system may be inoperable when irradiated fuel is in the fuel storage building, provided irradiated fuel is not being handled and neither the spent fuel cask nor the cask crane are moved over the spent fuel pit during the period of inoperability.

Basis

The equipment and general procedures to be utilized during refueling, fuel handling, and storage are discussed in the FSAR. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling, fuel handling, reactor maintenance or storage operations that would result in a hazard to public health and safety.(1) Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The residual heat removal pump is used to maintain a uniform boron concentration.

The shutdown margin indicated will keep the core subcritical, even if all control rods were withdrawn from the core. During refueling, the reactor refueling cavity is filled with approximately 342,000 gallons of water from the refueling water storage tank with a boron concentration of 2000 ppm. A shutdown margin of 10% $\Delta k/k$ in the cold condition with all rods inserted will also maintain the core subcritical even if no control rods were inserted into the reactor.(2) Periodic checks of refueling water boron concentration and residual heat removal pump operation insure the proper shutdown margin. The requirement for direct communications allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

In addition to the above safeguards, interlocks are utilized during refueling to ensure safe handling. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accomodate only one fuel assembly at a time.

The 120-hour decay time following the subcritical condition and the 23 feet of water above the top of the irradiated fuel assemblies are consistent with the assumptions used in the dose calculaion for the fuel-handling accident.

The waiting time of 400 hours required following plant shutdown before unloading more than one region of fuel from the reactor assures that the maximum pool water temperature will be within design objectives as stated in the FSAR.

The requirement for the fuel storage building emergency ventilation system to be operable is established in accordance with standard testing requirements to assure that the system will function to reduce the offsite dose to within acceptable limits in the event of a fuel-handling accident. The system is actuated upon receipt of a signal from the area high activity alarm or by a manually-operated switch. The system is tested prior to fuel handling and is in a standby basis.

When fuel in the reactor is moved before the reactor has been subcritical for at least 365 hours, the limitations on the containment vent and purge system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere.

The minimum spent fuel pit boron concentration and the 90-day restriction of the movement of the spent fuel cask to allow the irradiated fuel to decay were specified in order to minimize the consequences of an unlikely sideways cask drop.

When the spent fuel cask is being placed in or removed from its position in the spent fuel pit, mechanical stops incorporated in the bridge rails make it impossible for the bridge of the crane to travel further north than a point directly over the spot reserved for the cask in the pit. Thus, it will be possible to handle the spent fuel cask with the 40-ton hook and to move new fuel to the new fuel elevator with a 5-ton hook, but it will be impossible to carry any objective over the spent fuel storage area with either the 40 or 5-ton hook of the fuel storage building carne.

4.13 Containment Vent and Purge System

Applicability

This specification applies to the surveillance requirements of the containment vent and purge system when reactor fuel is anticipated to be moved before the reactor has been subcritical for at least 365 hours.

Objective

To verify the operability of the containment vent and purge system when reactor fuel is anticipated to be moved before the reactor has been subcritical for at least 365 hours.

Specification

If fuel movement is to take place before the reactor has been subcritical for at least 365 hours, the containment vent and purge system shall be demonstrated operable as follows:

a. Within 18 months prior to fuel movement and (1) after each complete or partial replacement of a HEPA filter or charcoal adsorber bank within 18 months prior to fuel movement, or (2) after structural maintenance on the HEPA filter or charcoal adsorber housing within 18 months prior to fuel movement, which could effect system operation:

   1. Verify that the charcoal adsorbers remove $\geq$99% of halogenated hydrocarbon refrigerant test gas when they are tested in-place while operating the ventilation system at the operating flow $\pm$ 10%.

   2. Verifying that the HEPA filter banks remove $\geq$99% of the DOP when they are tested in-place while operating the ventilation system at the operating flow rate $\pm$ 10%.

b. Within 18 months prior to fuel movement and after every 720 hours of system operation, subject a representative sample of carbon from the charcoal adsorbers to a laboratory analysis and verify within 31 days a removal efficiency of $\geq$90% for radioactive methyl iodine at an operating air flow velocity $\pm$20% per test 5.b in Table 2 of Regulatory Guide 1.52, March 1978.

Basis

The operability of this system and the resulting iodine removal capacity are consistent with accident analyses. The representative carbon sample will be two inches in diameter with a length equal to the thickness of the bed.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 30 TO FACILITY OPERATING LICENSE NO. DPR-64

POWER AUTHORITY OF THE STATE OF NEW YORK

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

Introduction

By letter dated January 17, 1977, the staff requested the Power Authority of the State of New York, Inc. (the licensee) to evaluate the previously unevaluated potential consequences of a postulated Fuel Handling Accident Inside Containment (FHAIC) at Indian Point Unit 3 (Indian Pt. 3). The licensee submitted, in a letter dated March 21, 1977, an evaluation of the FHAIC. The staff reviewed this submittal and requested in a letter dated May 5, 1977, that the licensee provide a basis for his model for mixing and for isolating the containment before a complete release of activity occurs. The staff also requested an analysis including the worst single failure during this accident. The licensee stated, by letter dated June 15, 1977, that the potential consequences for the worst single failure in the accident are 277.8 Rem Thyroid and 1.24 Rem Whole Body. The licensee described the assumptions used for containment mixing and locations of monitors which will automatically isolate the containment. The staff reviewed the licensee's June 15, 1977, submittal and concluded additional actions were needed to provide adequate assurance that the potential consequences of this accident were less than the guidelines of 10 CFR Part 100. By letter dated January 12, 1978, the staff proposed possible means to provide adequate assurance that the consequences of the FHAIC are within the guidelines of 10 CFR Part 100 for Indian Point 3. These proposals being: (1) increase the minimum time after shutdown before refueling, (2) redundant radiation monitors on the operating floor which will automatically isolate the containment, (3) a safety grade duct and charcoal filter on the purge exhaust from the containment, (4) smoke tests or other experiments or analysis which will demonstrate that the radioactivity released from the damaged fuel assembly would be mixed in the containment, or (5) conservative analysis which demonstrates that the containment would be isolated in a timely manner by the existing monitors assuming a single failure.

The licensee, in a letter dated February 14, 1978, agreed to increase the minimum time between shutdown and fuel handling from 100 hours to 120 hours in the Indian Point 3 Technical Specifications. This Technical Specification change was evaluated and approved in the safety evaluation dated March 22, 1978, for Indian Point 3. This change reduces the magnitude of radioactivity in the spent fuel assemblies available for release during this accident and provides additional assurance that the potential consequences are within the Part 100 guidelines.

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In addition, in our review of Unresolved Safety Issues at Indian Point 3, we noted that a change in the Technical Specifications was needed in Section 3.8 to add protection against potential heavy load drops on spent fuel. (See p. 27 of Enclosure 1 of our April 9, 1980 letter.)

By letter dated May 6, 1980, the licensee proposed changes to the Technical Specifications in response to our February 27, 1980 letter. These changes add surveillance requirements for the containment vent and purge system and add precautions for the control of heavy loads over the reactor when the vessel head is removed.

Evaluation

We have completed our review of the licensee's March 21, 1977, June 15, 1977, February 14, 1978 and May 6, 1980 submittals, which address the potential consequences of an accident involving spent fuel handling inside containment. We have performed an independent analysis of the FHAIC. Our assumptions and the resulting potential consequences at the Exclusion Area Boundary are given in Table 1.

We conclude that testing the Purge System charcoal filters every 720 hours of system operation for a 90% methyl iodine removal efficiency at 95% relative humidity will provide adequate assurances that the charcoal adsorber iodine removal efficiency is at least that which we have given in Table 1 and have assumed in our evaluation of the FHAIC. Based on the above discussed technical specifications on the charcoal adsorbers, degradation of the adsorbers during operation of the Containment Purge System (CPS) and a margin of safety to assure the charcoal radioiodine removal efficiencies are at least the efficiencies assumed in our evaluation of the FHAIC, we have assigned a 70% charcoal radioiodine removal efficiency for the CPS.

We conclude that the implementation of these Technical Specifications into the Indian Point 3 Technical Specifications will provide adequate assurance that the potential consequences of a postulated FHAIC are appropriately within the guidelines of 10 CFR Part 100. Appropriately within the guidelines of 10 CFR Part 100 has been defined as less than 100 Rem to the thyroid. This is based on the probability of this event relative to other events which are evaluated against 10 CFR Part 100 exposure guidelines. Whole body doses were also examined, but they are not controlling due to decay of the short-lived radioisotopes prior to fuel handling.

In our review, we did not require that the CPS be safety grade and did not consider the Single Failure Criteria, IEEE Standards, seismic design and equipment quality group classification. The CPS is not safety grade. We conclude that this is acceptable because the potential consequences of the postulated FHAIC are within the exposure guidelines of 10 CFR Part 100 with no

credit given for operation of the CPS. In addition, the surveillance requirements for the CPS filters discussed above are less than the requirements on safety grade ventilation filter systems because to have the potential consequences of this accident appropriately within the exposure guidelines of 10 CFR Part 100, more stringent surveillance requirements on the non-safety grade CPS filters are not needed.

A recent study[1] has indicated that dropping a spent fuel assembly into the core during refueling operations may potentially cause damage to more fuel pins than has been assumed for evaluating the FHAIC. This study has indicated that up to all of the fuel pins in two spent fuel assemblies, the one dropped and the one hit, may be damaged because of the embrittlement of fuel cladding material from radiation in the core. The radiation embrittlement would occur within the fuel's first few months of operations.

The probability of the postulated fuel handling accident inside containment is small. Not only have there been several hundred reactor-years of plant operating experience with only a few accidents involving spent fuel being dropped into the core, but none of these accidents has resulted in measurable releases of activity. The potential damage to spent fuel estimated by the study was based on the assumption that a spent fuel assembly falls about 14 feet directly onto one other assembly in the core; an impact which results in the greatest energy available for crushing the fuel pins in both assemblies. This type of impact is unlikely because the falling assembly would be subjected to drag forces in the water which should cause the assembly to skew out of a vertical fall path.

Based on the above, we have concluded that the likelihood of a spent fuel assembly falling into the core and damaging all the fuel pins in two assemblies is sufficiently small that refueling operations inside containment are not a safety concern which requires immediate remedial action. However, because there is a chance that more than one spent fuel assembly may be damaged during refueling, we are reviewing the study and the probability and consequences of dropping a spent fuel assembly in the core and damaging more fuel pins than the equivalent of one assembly. The objective of this review is to determine if any additional restrictions on fuel handling operations or plant operating procedures are needed. Any conclusions of this review which are applicable to this plant will be implemented.

We have calculated the potential radiological consequences of a fuel assembly drop onto the core assuming all the fuel pins in two spent fuel assemblies are ruptured. If, for both assemblies, we use the assumptions given in Regulatory Guide 1.25, and taking no credit for the non-ESF charcoal filters, the potential consequences of this accident are greater than the guidelines of 10 CFR Part 100. However, the source term defined in Regulatory Guide 1.25 is conservative because (1) these two assemblies are unlikely to both have the high power

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[1] M. Singh, "Fuel Assembly Handling Accident Analysis," EG&G Idaho Technical Report RE-A-78-227, October 1978.

peaking factor and clad gap activity used in Regulatory Guide 1.25, and (2) the pool decontamination factor for inorganic iodine may well be greater than that used in Regulatory Guide 1.25. Taking into account more realistic values for power peaking factor, clad gap activity and pool decontamination, we conclude that potential consequences of this postulated accident should not be greater than the exposure guidelines of 10 CFR Part 100. Because these potential consequences are less than the guidelines of 10 CFR Part 100, we have concluded that no additional restrictions on fuel handling operations and plant operating procedures are needed while our review is underway.

The results of this analysis warranted an investigation of a similar accident in the spent fuel pool even though it is located outside containment. For this, a drop of 2-12 feet was postulated and the analysis performed in the same manner as previously described. Results indicate that in this scenario damage to the missile or target is minimal. No fuel pins in either fuel assembly were calculated to be ruptured.

After performing an independent analysis of the radiological consequences of an FHAIC to any individual located at the nearest exclusion boundary, the staff concludes that the doses for one assembly failure are appropriately within the guideline values of 10 CFR Part 100 and for failure of two assemblies are within the guideline values of 10 CFR Part 100 and are, therefore, acceptable.

We also conclude that the proposed change to the Technical Specifications which adds precautions for the control of heavy loads over the reactor when the vessel head is removed is acceptable. With this change, we find that Indian Point 3 has substantial protection against potential heavy load drops on spent fuel in the reactor vessel and that there is reasonable assurance that the health and safety of the public is protected while the generic task "Control of Heavy Loads Near Spent Fuel" is completed and its results implemented.

Environmental Consideration

We have determined that the 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 impact 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.

Date: June 13, 1980

Table 1

ASSUMPTIONS FOR AND POTENTIAL CONSEQUENCES OF THE POSTULATED FUEL HANDLING ACCIDENTS AT THE EXCLUSION AREA BOUNDARY FOR INDIAN POINT STATION UNIT 3

Assumptions:

Guidance in Regulatory Guide 1.25	
Power Level	3025 MWt
Fuel Exposure Time	3 years
Power Peaking Factor	1.65
Equivalent Number of Assemblies Damaged	1
Number of Assemblies in Core	193
Charcoal Filter Efficiency Elemental and Organic	70 percent
Decay time before moving fuel	120 hours
0-2 hours, X/Q Value, Exclusion Area Boundary (ground level release)	1.1×10^{-3} sec/m^3

	Doses, Rem	
	Thyroid	Whole Body
Exclusion Area Boundary (EAB) Consequences from Accidents Inside Containment	73	.83

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-286

POWER AUTHORITY OF THE STATE OF NEW YORK

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 30 to Facility Operating License No. DPR-64, issued to the Power Authority of the State of New York (the licensee), which revised Technical Specifications for operation of the Indian Point Nuclear Generating Unit No. 3 (the facility) located in Buchanan, Westchester County, New York. The amendment is effective as of the date of issuance.

The amendment changes the Technical Specifications by adding surveillance requirements for the containment vent and purge system and by adding precautions for the control of heavy loads over the reactor when the vessel head is removed.

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.

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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 May 6, 1980, (2) Amendment No. 30 to License No. DPR-64, 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 White Plains Public Library, 100 Martine Avenue, White Plains, New York. 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 Licensing.

Dated at Bethesda, Maryland, this 13th day of June, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing