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Docket No. 50-286

Mr. George T. Berry
General Manager and Chief Engineer
Power Authority of the State of New York
10 Columbus Circle
New York, New York 10019

Dear Mr. Berry:

In response to your application dated April 26, 1979, the Commission has issued the enclosed Amendment No. 26 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3.

The amendment revises the Technical Specifications to require actuation of safety injection based on 2 out of 3 channels of low pressurizer pressure.

As discussed with you, it is understood and acceptable for the pressurizer level bistables to be returned to their normal position while this modification is underway. Further, it is understood that the changes will be made one train at a time, with each train tested before being placed into service.

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

Sincerely,

Original Signed By

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 26 to DPR-64
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

TAC# 11669

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DATE	05/18/79	06/1/79	05/18/79	05/31/79	05/31/79	05/30/79

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

May 31, 1979

Docket No. 50-286

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General Manager and Chief Engineer
Power Authority of the State of New York
10 Columbus Circle
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Sincerely,

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
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cc: w/enclosures
See next page

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Mayor, Village of Buchanan
188 Westchester Avenue
Buchanan, New York 10511

cc: Director, Technical Assessment Division
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Region II Office
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26 Federal Plaza
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 NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 26
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 April 26, 1979, 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 regulations;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility License No. DPR-64 is hereby amended to read as follows:

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"(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 26 , 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

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 31, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 26

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Replace the following pages of the Technical Specifications contained in Appendix A of the above-indicated license with the attached pages bearing the same numbers, except as otherwise indicated. The changed areas on the revised pages are reflected by a marginal line.

Remove	Insert
3.5-1 through 3.5-6	3.5-1 through 3.5-6
Table 3.5-1	Table 3.5-1
Table 3.5-2	Table 3.5-2
Table 3.5-3	Table 3.5-3
Table 3.5-4 (Sheet 1)	Table 3.5-4 (Sheet 1)

3.5 INSTRUMENTATION SYSTEMS

Operational Safety Instrumentation

Applicability

Applies to plant instrumentation systems.

Objectives

To provide for automatic initiation of the Engineered Safety Features in the event that principal process variable limits are exceeded, and to delineate the conditions of the plant instrumentation and safety circuits necessary to ensure reactor safety.

Specification

3.5.1 When the plant is not in the cold shutdown condition, the Engineered Safety Features initiation instrumentation setting limits shall be as stated in Table 3.5-1.

3.5.2 For instrumentation testing or instrumentation channel failure, plant operation shall be permitted to continue in accordance with Tables 3.5-2 through 3.5-4. No more than one channel of a particular protection channel set shall be tested at the same time. By definition, an instrumentation channel failure shall not be regarded as a channel being tested.

3.5.3 In the event the number of in-service channels of a particular function is less than the minimum number of Operable Channels (Col. 3), or the Minimum Degree of Redundancy (Col. 4) cannot be achieved, operation shall be limited according to the requirement shown in Column 5 of Tables 3.5-2 through 3.5-4.

3.5.4 In the event of instrumentation channel failure permitted by specification 3.5.2, the Minimum Degree of Redundancy listed in Tables 3.5-2 through 3.5-4 may be reduced by one, but to not less than zero, and the Minimum Number of Operable Channels listed in these tables may be reduced by one, but not to less than one (except as noted in Table 3.5-3) for a period of 8 hours while instrument channels are tested. The failed channel may be blocked to prevent an unnecessary reactor trip during this time. In the case of three loop operation, the out-of-service channel is permitted to be bypassed during the test period.

3.5.5 The low pressurizer pressure safety injection trip shall be unblocked when the pressurizer pressure is $\geq$ 2000 psig.

3.5.6 At least one source range and one intermediate range nuclear instrument channel shall be operable prior to reactor start-up.

3.5.7 When the reactor is not in the cold shutdown condition, the instrumentation requirements as stated in Table 3.5-5 shall be met.

Basis

Instrumentation has been provided to sense accident conditions and to initiate operation of the Engineered Safety Features[1].

Safety Injection System Actuation

Protection against a Loss of Coolant or Steam Break accident is brought about by automatic actuation of the Safety Injection System which provides emergency cooling and reduction of reactivity.

The Loss of Coolant Accident is characterized by depressurization of the Reactor Coolant System and rapid loss of reactor coolant to the containment. The Engineered Safety Features have been designed to sense the effects of the Loss of Coolant accident by detecting low pressurizer pressure and generate signals actuating the SIS active phase based upon these signals. The SIS active phase is also actuated by a high containment pressure signal (Hi-Level) brought about by loss of high enthalpy coolant to the containment. This actuation signal acts as a backup to the low pressurizer pressure signal actuation of the SIS and also adds diversity to protection against loss of coolant.

Signals are also provided to actuate the SIS upon sensing the effects of a steam line break accident. Therefore, SIS actuation following a steam line break is designed to occur upon sensing high differential steam pressure between any two steam generators or upon sensing high steam line flow in coincidence with low reactor coolant average temperature or low steam line pressure.

The increase in the extraction of RCS heat following a steam line break results in reactor coolant temperature and pressure reduction. For this reason protection against a steam line break accident is also provided by low pressurizer pressure signals actuating safety injection.

Protection is also provided for a steam line break in the containment by actuation of SIS upon sensing high containment pressure.

SIS actuation injects highly borated fluid into the Reactor Coolant System in order to counter the reactivity insertion brought about by cooldown of the reactor coolant which occurs during a steam line break accident.

Containment Spray

The Engineered Safety Features actuation system also initiates containment spray upon sensing a high containment pressure signal (Hi-Hi Level). The containment spray acts to reduce containment pressure in the event of a loss of coolant or steam line break accident inside the containment. The spray cools the containment directly and limits the release of fission products by absorbing iodine should it be released to the containment.

Containment spray is designed to be actuated at a higher containment pressure (∿ 50% of containment design pressure) than the SIS (∿ 10% of containment design pressure). Since spurious actuation of containment spray is to be avoided, it is automatically initiated only on coincidence of Hi-Hi Level containment pressure sensed by both sets of two-out-of-three containment pressure signals and coincidence with the S.I. Signal.

Steam Line Isolation

Steam line isolation signals are initiated by the Engineered Safety Features closing all steam line stop valves. In the event of a steam line break, this action prevents continuous, uncontrolled steam release from more than one steam generator by isolating the steam lines on high containment pressure (Hi-Hi Level) or high steam line flow. Protection is afforded for breaks inside or outside the containment even when it is assumed that there is a single failure in the steam line isolation system.

Feedwater Line Isolation

The feedwater lines are isolated upon actuation of the Safety Injection System in order to prevent excessive cooldown of the reactor coolant system. This mitigates the effect of an accident such as steam break which in itself causes excessive coolant temperature cooldown.

Feedwater line isolation also reduces the consequences of a steam line break inside the containment, by stopping the entry of feedwater.

Setting Limits

1. The Hi-Level containment pressure limit is set at about 10% of containment design pressure. Initiation of Safety Injection protects against loss of coolant[2] or steam line break[3] accidents as discussed in the safety analysis.

2. The Hi-Hi Level containment pressure limit is set at about 50% of containment design pressure. Initiation of Containment Spray and Steam Line Isolation protects against large loss of coolant[2] or steam line break accidents[3] as discussed in the safety analysis.

3. The pressurizer low pressure limit is set substantially below system operating pressure limits. However, it is sufficiently high to protect against a loss of coolant accident as shown in the safety analysis[2]. The trip is bypassed below 2000 psig to prevent inadvertent actuation of the Engineered Safeguards when the reactor is shutdown.

TABLE 3.5-1

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

No.	FUNCTIONAL UNIT	CHANNEL	SETTING LIMIT
1.	High Containment Pressure (Hi Level)	Safety Injection	$\leq$ 3.5 psig
2.	High Containment Pressure (Hi-Hi Level)	a. Containment Spray b. Steam Line Isolation	$\leq$ 23 psig
3.	Pressurizer Low Pressure	Safety Injection	$\geq$ 1700 psig
4.	High Differential Pressure Between Steam Lines	Safety Injection	$\leq$ 150 psi
5.	High Steam Flow in 2/4 Steam Lines Coincident with Low T_{avg} or Low Steam Line Pressure	a. Safety Injection b. Steam Line Isolation	$\leq$ 40% of full steam flow at zero load $\leq$ 40% of full steam flow at 20% load $\leq$ 110% of full steam flow at full load $\geq$ 540°F T_{avg} $\geq$ 600 psig steam line pressure

TABLE 3.5-2 (Sheet 1 of 2)

REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. NUMBER OF OPERABLE CHANNELS	4 MIN. DEGREE OF REDUNDNACY	5 OPERATOR ACTION IF CONDITIONS OF COL. 3 OR 4 CANNOT BE MET*
1.	Manual Reactor Yrip	2	1	1	0	Maintain hot shutdown
2.	Nuclear Flux Power Range	4	2	3	2	Maintain hot shutdown
		4	2	2	1	For zero power physics tests only
3.	Overtemperature ΔT	4	2	3	2	Maintain hot shutdown
4.	Overpower ΔT	4	2	3	2	Maintain hot shutdown
5.	Low Pressurizer Pressure	4	2	3	2	Maintain hot shutdown
6.	Hi Pressurizer Pressure	3	2	2	1	Maintain hot shutdown
7.	Pressurizer-Hi Water Level	3	2	2	1	Maintain hot shutdown
8.	Low Flow One Loop (Power ≥ P-8)	3/loop	2/loop (any loop)	2/operable loop	1/operable loop	Maintain hot shutdown
	Low Flow Two Loops (Power < P-8 and ≥ P-10)	3/loop	2/loop (any two loops)	2/operable loop	1/operable loop	Maintain hot shutdown

TABLE 3.5-2 (Sheet 2 of 2)

		1	2	3	4	5
9.	Lo Lo Steam Generator Water Level	3/loop	2/loop	2/loop	1/loop	Maintain hot shutdown
10.	Undervoltage 6.9 KV Bus	1/bus	2	3	2	Maintain hot shutdown
11.	Low Frequency 6.9 KV Bus**	1/bus	2	3	2	Maintain hot shutdown
12.	Turbine trip (electrical over-speed protection)	3	2	2	1	Turbine shutdown (turbine stop valves closed)

* Maintain hot shutdown means maintain or proceed to hot shutdown within 4 hours using normal operating procedures, if the unacceptable condition arises during operation.

** 2/4 trips all four reactor coolant pumps.

TABLE 3.5-3 (Sheet 1 of 2)

INSTRUMENTATION OPERATING CONDITION FOR ENGINEERED SAFETY FEATURES

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. NUMBER OF OPERABLE CHANNELS	4 MIN. DEGREE OF REDUNDANCY	5 OPERATOR ACTION IF CONDITIONS OF COL. 3 OR 4 CANNOT BE MET*****
1.	SAFETY INJECTION					
	a. Manual	2	1	1	0	Cold Shutdown
	b. High Containment Pressure (Hi Level)	3	2	2	1	Cold Shutdown
	c. High Differential Pressure Between Steam Lines	3/steam line	2/steam line	2/steam line	1/steam line	Cold Shutdown
	d. Pressurizer Low Pressure	3	2	2	1	Cold Shutdown
	e. High Steam Flow in 2/4 Steam Lines Coincident with Low Tavg or Low Steam Line Pressure	2/steam line	1/2 in any 2 steam lines	2 channels in each of 3 steam lines	2	Cold Shutdown or main steam isolation valves closed
		4 Tavg Signals	2	3	2	
		4 Pressure Signals	2	3	2	
	f. Pressurizer Low Pressure and (Automatic Unblock)	3	2	2****	1****	Cold Shutdown

TABLE 3.5-3 (Sheet 2 of 2)

	1	2	3	4	5
2. CONTAINMENT SPRAY					
a. Manual	2	2	2	0***	Cold Shutdown
b. High Containment Pressure (Hi Hi Level)	2 sets of 3	2 of 3 in each set	2 per set	1/set	Cold Shutdown

* Permissible to bypass if reactor coolant pressure less than 2000 psig.

*** Must actuate 2 switches simultaneously.

**** The Minimum Number of Operable Channels and the Minimum Degree of Redundancy may be reduced to zero if the SI bypass is in the unblocked position.

***** If the condition of Column 3 or 4 cannot be met, the reactor shall be placed in the hot shutdown condition, utilizing normal operating procedures, within 4 hours of the occurrence. If the conditions are not met within 24 hours of the occurrence, the reactor shall be placed in the cold shutdown condition, or the alternate condition, if applicable, within an additional 24 hours.

TABLE 3.5-4 (Sheet 1 of 2)

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MIN. DEGREE OF REDUN-ANCY	5 OPERATOR ACTIO IF CONDITIONS COLUMN 3 OR 4 CANNOT BE MET*
1.	CONTAINMENT ISOLATION					
	a. Automatic Safety Injection (Phase A)	See Item No. 1(b) of Table 3.5-3				Cold Shutdown
	b. Containment Pressure (Phase B)	See Item No. 2(b) of Table 3.5-3				Cold Shutdown
	c. Manual					
	Phase A	2	1	1	0	Cold Shutdown
	Phase B	See Item 2(a) of Table 3.5-3				Cold Shutdown
2.	STEAM LINE ISOLATION					
	a. High Steam Flow in 2/4 Steam Lines Coincident with Low T_{avg} or Low Steam Line Pressure	See Item No. 1(e) of Table 3.5-3				Cold Shutdown o Main Steam Isol Valves Closed
	b. High Containment Pressure (Hi Hi Level)	See Item No. 2(b) of Table 3.5-3				Cold Shutdown o Main Steam Isol Valves Closed**
	c. Manual	1/loop	1/loop	1/loop	0	Cold Shutdown o Main Steam Isol Valves Closed

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 26 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 April 26, 1979, Power Authority of the State of New York (the licensee) requested an amendment to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. The proposed amendment would change the Technical Specifications to require actuation of safety injection based on 2 out of 3 channels of low pressurizer pressure. Until this change has been accomplished, it is understood that the low pressurizer level trips in the existing safety injection logic would remain tripped as provided for in IE Bulletin 79-06A (Item 3) and 79-06A, Revision 1.

As discussed with you, it is further understood and acceptable for the pressurizer level bistables to be returned to their normal position while this modification is underway. Further, it is understood that the changes will be made one train at a time, with each train tested before being placed into service.

Discussion

As a result of our ongoing review of the events associated with the March 28 accident at Three Mile Island Unit 2, the NRC Office of Inspection and Enforcement issued a number of IE Bulletins describing actions to be taken by licensees. IE Bulletin 79-06 (April 11, 1979) called for licensees with Westinghouse PWR's to instruct operators to manually initiate safety injection when the pressure indication reaches the actuation setpoint whether or not the level indication has dropped to the actuation setpoint. IE Bulletin 79-06A (April 14, 1979) further called for these licensees to trip the low pressurizer level bistables such that, when the pressurizer pressure reaches the low setpoint safety injection would be initiated regardless of the pressurizer level. This action was completed on April 14, 1979.

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IE Bulletin 79-06A, Revision 1 (April 18, 1979) modified the action called for in 79-06A by allowing pressurizer level bistables to be returned to their normal (untripped) operating positions during the pressurizer pressure channel functional surveillance tests.

The effect of tripping the pressurizer low level bistables which are normally coincident with the pressurizer low pressure bistables, has the effect of reducing this safety injection actuation logic to a 1 out of 3 logic. A single instrument failure of 1 of the 3 low pressure bistable channels could therefore result in an unwanted safety injection. To prevent this, the licensee proposed in an April 26, 1979 letter, a design modification which would align the existing pressurizer low pressure bistables in a 2 out of 3 logic.

Evaluation

The proposed modification to the safety injection actuation system entails removing the pressurizer level signal from each of the pressurizer level/pressure channel trip and converting the system to a two-out-of-three pressurizer low pressure trip. The instrumentation logic takes pressurizer pressure signals from three pressure transmitters and initiates a safety injection actuation whenever two of the three signals reach the low pressure setpoint of 1700 psig. These modifications will satisfy the requirements of IEEE 279-1971, and other standards of installation required during the plant construction stage. We find these modifications acceptable.

We have reviewed the instrumentation and controls aspect of the proposed change in accordance with IEEE-279 and other applicable standards and Reg. Guides. The modification eliminates pressurizer level as a required initiating signal to actuate ECCS. The licensee proposes to use a 2-out-of-3 logic on low pressurizer pressure alone. Separation of trains will be maintained, testability will be maintained, and verification of proper actuation of the first train can be performed prior to modification of the second train.

We have reviewed the instrumentation power sources and determined there are four 120V instrument buses. Three of the buses (31, 32 and 33) are supplied from inverters which in turn are energized from three independent battery banks for the plant. The fourth bus (34) is supplied by a constant voltage transformer connected to 480 volt (emergency power) motor control center 36B. Alternate power to the four buses can be supplied from the lighting switchgear. The three pressurizer pressure transmitters channels are energized from buses 31, 32 and 34; consequently, a single failure will fail only one instrument channel. The licensee has committeed to install a fourth battery and inverter as an additional source of instrument power during the next refueling outage (fall 1979), pending the receipt on site of the necessary materials, but to be installed not later than 1981 refueling outage.

The proposed Technical Specifications change Section 3.5 to specify automatic safety injection actuation on a two-out-of-three pressurizer low pressure of 1700 psig. We find the changes to the Technical Specifications to be acceptable.

Based on our review of the licensee's submittal, we conclude that the modifications to the safety injection actuation system logic satisfy the requirements of IEEE 279-1971 and that the associated Technical Specificaions are correct; and therefore, are acceptable.

We also conclude that the proposed change will be in accordance with the above standards and guides, and that none of the transient and accident analyses are adversely affected by the change. The only effect may be a sooner SI actuation. Therefore, we find the proposed change to be acceptable.

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 these amendments.

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: May 31, 1979

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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.26 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 revises the Technical Specifications to require actuation of safety injection based on 2 out of 3 channels of low pressurizer pressure.

The application for 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 amendments. 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 April 26, 1979, (2) Amendment No. 26 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, NW, Washington, D. C. and at the White Plains Public Library, 100 Martine Avenue, White Plains, New York. A single 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 31st day of May, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors