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Docket No. 50-247

Mr. John D. O'Toole
Vice President
Nuclear Engineering and Quality Assurance
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Dear Mr. O'Toole:

The Commission has issued the enclosed Amendment No. 79 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated August 11, 1980.

The amendment revises your Technical Specifications to require certain condensate valves to be open when the plant is above 350°F and provide for testing of the steam generator low level AFWS automatic actuation logic. During our review of your proposal, we found that certain changes were necessary. Your staff has agreed to these changes and they have been incorporated.

The accompanying Safety Evaluation completes our review of your auxiliary feedwater system. It serves to resolve Item F.5. of Appendix A of our February 11, 1980 Order and Items II.E.1.1 and II.E.1.2 of NUREG-0737 for Indian Point Unit 2.

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

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FR NOTICE + AMENDMENT

Enclosures:
1. Amendment No. 79 to DPR-26
2. Safety Evaluation
3. Notice of Issuance

Received 9-22-82

cc: w/enclosures
See next page

OFFICE	ORB#1:DL	ORB#1:DL	ORB#1:DL	ORB#4:DL	ORAB	AD:OR:DL	OELD
SURNAME	CParrish	JHannon/pn	SVarga	PWagner	TAlexion	GLainas	M.Karman
DATE	8/13/82	8/16/82	8/ /82	8/16/82	8/16/82	8/ /82	8/24/82

NRC FORM 318 (10-80) NRCM 0240 **OFFICIAL RECORD COPY** USGPO: 1981—335-960

Mr. John D. O'Toole
Consolidated Edison Company of New York, Inc.

cc: White Plains Public Library
100 Martine Avenue
White Plains, New York 10601

Joseph D. Block, Esquire
Executive Vice President
Administrative
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Louis F. Liberatori
Nuclear Licensing Engineer
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Ms. Ellyn Weiss
Sheldon, Harmon and Weiss
1725 I Street, N.W., Suite 506
Washington, D. C. 20006

Dr. Lawrence R. Quarles
Apartment 51
Kendal at Longwood
Kennett Square, Pennsylvania 19348

Theodore A. Rebelowski
Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 38
Buchanan, New York 10511

Brent L. Brandenburg
Assistant General Counsel
Consolidated Edison Company
of New York, Inc.
4 Irving Place - 1822
New York, New York 10003

Carl R. D'Alvia, Esquire
Attorney for the Village of
Buchanan, New York
395 South Riverside Avenue
Croton-on-Hudson, New York 10520

Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, New York 10511

Director, Technical Development
Programs
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Regional Radiation Representative
EPA Region II
26 Federal Plaza
New York, New York 10007

Thomas J. Farrelly, Esquire
Law Department
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Joan Holt, Project Director
New York Public Interest
Research Group, Inc.
5 Beekman Street
New York, New York 10038

Ronald C. Haynes
Regional Administrator - Region I
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Mr. John D. O'Toole
Consolidated Edison Company of New York, Inc.

cc: Mr. Charles W. Jackson
Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenues
Buchanan, New York 10511

Mr. Kevin Burke
Director, Regulatory Affairs
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenues
Buchanan, New York 10511

Mr. Frank Matra
Resident Construction Manager
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenues
Buchanan, New York 10511

Ezra I. Bialik
Assistant Attorney General
Environmental Protection Bureau
New York State Department of Law
2 World Trade Center
New York, New York 10047

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

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 79
License No. DPR-26

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

   A. The application for amendment by Consolidated Edison Company of New York, Inc, (the licensee) dated August 11, 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.

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 Operating License No. DPR-26 is hereby amended to read as follows:

    (2) <u>Technical Specifications</u>

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 79, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective 21 days from 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: August 30, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 79

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.4-1	3.4-1
-	3.4-1(a)
Table 4.1-1 (continued)	Table 4.1-1 (continued)

3.4 STEAM and POWER CONVERSION SYSTEM

Applicability

Applies to the operating status of the Steam and Power Conversion System.

Objective

To define conditions of the turbine cycle steam-relieving capacity. Auxiliary Feedwater System and City Water System operation is necessary to ensure the capability to remove decay heat from the core.

Specification

A. The reactor shall not be heated above 350°F unless the following conditions are met:

(1) A minimum ASME code approved steam-relieving capability of twenty (20) main steam valves shall be operable (except for testing).

(2) Three auxiliary feedwater pumps each capable of pumping a minimum of 400 gpm must be operable.

(3) A minimum of 360,000 gallons of water in the condensate storage tank and a backup supply from the city water supply.

(4) Required system piping, valves, and instrumentation directly associated with the above components operable.

(5) The main steam stop valves are operable and capable of closing in five seconds or less.

(6) The total iodine activity of I-131 and I-133 on the secondary side of the steam generator shall be less than or equal to 0.15 uCi/cc.

B. Except as modified by 3.4.C below, if any of the conditions of 3.4.A above can not be met within 72 hours, the reactor shall be placed in the hot shutdown condition within the next 12 hours and subsequently cooled below 350°F using normal operating procedures.

C. If when above 350°F one or both of the series valves (CT-6 and/or CT-64) in the condensate storage tank discharge line is closed, then:

(1) Immediately place the auxiliary feedwater pump controls in the manual mode, and

(2) Within one (1) hour, either the valve(s) shall be reopened or the valves from the alternate city water supply shall be opened and the auxiliary feedwater pump controls restored to the automatic mode.

If these requirements cannot be met, then:

(1) maintain the plant in a safe stable mode which minimizes the potential for a reactor trip, and

(2) continue efforts to restore water supply to the auxiliary feedwater system, and

(3) notify the NRC within 24 hours regarding the planned corrective action.

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
b. Low-Low Level AFWS Automatic Actuation Logic	N.A.	N.A.	M **	Test one logic channel per month on an alternating basis.
c. Station Blackout (Undervoltage)	N.A.	R	R	
d. Trip of Main Feedwater Pumps	N.A.	N.A.	R	
31. Reactor Coolant System Subcooling Margin Monitor	M	R	N.A.	
32. PORV Position Indicator (Limit Switch)	M	R	R	
33. PORV Block Valve Position Indicator (Limit Switch)	M*	R	R	
34. Safety Valve Position Indicator (Acoustic Monitor)	M	R	R	
35. Auxiliary Feedwater Flow Rate	M	R	R	
36. PORV Actuation/ Reclosure Setpoints	N.A.	R	N.A.	

*Except when valve operator is deenergized in accordance with specification 3.1.A.4.c
**To be effective upon start-up for cycle 6 operations.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 79 TO FACILITY OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

Introduction and Background

The Three Mile Island Unit 2 (TMI-2) accident and subsequent investigations and studies highlighted the importance of the Auxiliary Feedwater System (AFWS) in the mitigation of transients and accidents. As part of our assessment of the TMI-2 accident and related implications for operating plants, we evaluated the AFW systems for all operating plants having nuclear steam supply systems (NSSS) designed by Westinghouse (NUREG-0611) or Combustion Engineering (NUREG-0635). Our evaluations of these system designs are contained in the NUREGs along with our recommendations for each plant and the concerns which led to each recommendation. The objectives of the evaluation were to: (1) identify necessary changes in AFW system design or related procedures at the operating facilities in order to assure the continued safe operation of these plants, and (2) to identify other system characteristics of the AFW systems which, on a long term basis, may require system modifications. To accomplish these objectives, we:

(1) Reviewed plant specific AFW system designs in light of current regulatory requirements (SRP) and,

(2) Assessed the relative reliability of the various AFW systems under various loss of feedwater transients (one of which was the initiating event of TMI-2) and other posutlated failure conditions by determining the potential for AFW system failure due to common causes, single point vulnerabilities, and human error.

We concluded that the implementation of the recommendations identified during this review will considerably improve the reliability of the AFW systems for each operating plant.

Our original request for information was sent to Consolidated Edison of New York, Inc. on November 7, 1979. On June 13, 1980, the NRC issued an interim Safety Evaluation of the Auxiliary Feedwater (AFW) system reliability for Indian Point Unit No. 2. Our interim Safety Evaluation resolved all but six items of this issue. Specifically, in order to complete our review we requested information or submittals for the following items:

1. Technical Specifications for the valves in the single line from the Condensate Storage Tank.

2. Basis for AFW system flow requirements,

3. Results of the endurance tests on the AFW pumps,

4. Results of the evaluation of the capability of the present design to withstand internally generated missiles,

5. Design of the AFW system automatic initiation signals, and

6. Design of the independent power supplies to the valve position controllers.

Consolidated Edison of New York, Inc. responded to our request by letters dated June 30, 1980, August 11, 1980 (2 letters), November 26, 1980, February 26, 1981, and May 12, 1981.

This Safety Evaluation reviews the remaining open items.

Evaluation

1. Condensate Storage Tank Line Valves

   In the staff Safety Evlauation of June 13, 1980, we found the system proposed by the licensee acceptable provided appropriate technical specifications were proposed to insure that a proper flow path existed. Essentially this required certain condensate valves to be open when the system is required to be operational. The proposed technical specifications satisfy the intent of our interim Safety Evaluation. They require prompt action (but not immediate shutdown) if the condensate valves are in an improper lineup. They also require notification to the NRC if the system cannot be repaired within 24 hours. The proposed technical specifications are acceptable in this area.

2. Basis for AFW System Flow Requirements

   On August 11, 1980 the licensee submitted an analysis indicating that the AFW system must have a capacity to supply at least 400 gpm total to at least two steam generators. The Indian Point 2 AFW control valves are preset to deliver 150 gpm maximum to each steam generator (600 gpm total).

However, it is possible for a single failure to reduce total flow below the minimum capacity assumed in their analysis. Although operator action could increase flow by opening the control valves, the staff does not recognize operator action for at least 10 minutes following an accident or transient. On November 26, 1980 the licensee submitted the results of an analysis which demonstrated that without operator action for 10 minutes and assuming the limiting failure, the reduced flow rate will still adequately remove decay heat and meet the criteria used for the August 11, 1980 analysis. Therefore the staff concludes that the basis for the Indian Point 2 AFW system flow requirements is acceptable.

3. Results of the Endurance Tests on the AFW Pumps

On June 30, 1980 the licensee submitted results of endurance testing of the auxiliary feedwater pumps. The tests were conducted in accordance with staff criteria. The test results were satisfactory and staff concerns in this area are resolved.

4. Missile Analysis

On August 11, 1980 the licensee submitted a missile analysis which determined that the auxiliary feedwater pump turbine was the only potential source of an internally generated missile which could damage vital equipment enough to disable the AFW system. Four options were being considered as corrective measures. By letter dated February 26, 1981 the licensee informed the NRC that the option chosen was to install a protective missile shield around the turbine for protection against missiles generated at destructive overspeed. By letter dated May 12, 1981, the licensee informed the NRC that the shield was installed. This action resolved the staff concern in this area.

5. Safety Grade Requirements for the AFW System

To improve the reliability of the Auxiliary Feedwater System (AFWS), the staff is requiring licensees to upgrade the system when necessary to ensure the timely automatic initiation when required. The system upgrade was to proceed in two phases. In the short term, as a minimum, control grade signals and circuits were to be used to automatically initiate the AFWS. This control grade system was required to meet the following requirements from NUREG-0578, Section 2.1.7.a.

a. The design shall provide for the automatic initiation of the auxiliary feedwater system.

b. The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

c. Testability of the initiating signals and circuits shall be a feature of the design.

d. The initiating signals and circuits shall be powered from the emergency buses.

e. Manual capability to initiate the auxiliary feedwater system from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.

f. The a-c motor-driven pumps and valves in the auxiliary feedwater system shall be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

g. The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room.

In the long term, these signals and circuits were to be upgraded in accordance with safety grade requirements. Specifically in addition to the above requirements, the automatic initiation signals and circuits must have independent channels, use qualified components, have system bypassed/inoperable status features and conform to control system interaction criteria, as stipulated in IEEE Standard 279. The staff review in this area included the last two open items in our interim Safety Evaluation of June 13, 1980.

In order to improve the operators' ability to ascertain from the control room the performance of the auxiliary feedwater system (AFWS) for various transient/accident conditions, the staff is requiring licensees to install safety grade AFW flow channels or upgrade existing channels to comply with safety grade requirements. Specifically, these instrument channels must be powered from the vital (battery backed) instrument buses, testability of these channels must be a feature of the design, and the instrumentation indicating the performance of the AFWS (flow and wide range level indication for each steam generator) must satisfy the single failure criterion. In addition, these channels must be independent, environmentally qualified, and conform to control system interaction criteria as stipulated in IEEE Standard 279.

The auxiliary feedwater system (AFWS) installed at the Indian Point Unit 2 (IP-2) Nuclear Plant utilizes two electric motor driven pumps and a single turbine driven pump. Each of the two motor driven pumps supply two of the four steam generators and start automatically on low-low water level in any steam generator (two-out-of-three level transmitters per steam generators), automatic trip of either main feed pump, loss of offsite power concurrent with a unit trip, or a safety injection signal. The turbine driven pump will automatically start on either a low-low water level in any two of the four steam generators (two-out-of-three level transmitters for each generator) or a loss of offsite power concurrent with a unit trip.

The automatic initiation signals and circuits for the AFWS at Indian Point Unit 2 comply with the single failure criterion of IEEE Standard 279. The motor driven AFW pumps are tested monthly and the steam turbine driven pump is tested at intervals not greater than six months. The analog and logic channels for steam generator low-low water level and safety injection are tested monthly. In addition, all AFWS automatic initiation signals will be tested at refueling intervals by injecting a simulated actuation signal into the initiation circuitry being tested and verifying that all appropriate actions take place, including starting of the pump(s).

The automatic initiation signals and associated circuitry used to actuate the auxiliary feedwater system are powered from the emergency buses. The redundant channels which provide these signals are physically separated and electrically independent. The ac motor driven pumps and valves in the IP-2 AFWS are powered from the emergency buses and are included in the automatic sequencing of loads onto these buses.

No single failure within the manual or automatic initiation systems for the IP-2 auxiliary feedwater system will prevent initiation of the system by automatic or manual means, respectively. Auxiliary feedwater flow to all four steam generators is supplied through individual feed regulating valves which can be manually controlled from the main control board or local to the valve. A steam generator high water level alarm from the narrow range level instrument channel is provided in the control room to aid the operator in this control function.

The individual feedwater regulating valves associated with the motor driven AFW pumps are normally closed. However, these valves are designed to go to the 35% open position after the AFW pumps have started and developed discharge pressure as sensed by two pressure transmitters (PT406A and 406B), one transmitter per motor driven AFW pump. Each transmitter sends signals to open two of the four discharge valves. Since the four discharge valves from the turbine driven AFW pump are kept closed and do not receive open signals on a pump start, the staff became concerned that a single failure of one of the two pressure transmitters might prevent adequate AFW flow when needed. Specifically, a steam line break coupled with a failure of the pressure transmitter associated with the motor driven pump not feeding the depressurized steam generator could prevent sufficient AFW flow. All discharge valves to two of the three intact steam generators would remain closed and the flow to the third intact generator would most likely be forced into the depressurized generator. The pressure transmitter would have to fail low (i.e., little or no discharge pressure). To get flow to the intact steam generators, the discharge valves from the turbine driven AFW pump would have to be manually opened from the control room. These valves are normally kept closed due to water hammer considerations.

The staff has determined that sufficient time exists for operator action to properly distribute auxiliary feedwater flow following a break. Therefore, the staff concludes that failure of a pressure transmitter will not result in unacceptable consequences. In addition, the AFWS flow indication installed at Indian Point Unit 2 complies with the long term requirements of NUREG-0737 Section II.E.1.2.2 and will indicate a no flow conditions in the event of a transmitter failure.

There are no operational bypasses associated with the IP-2 AFWS. If part of the AFWS has been bypassed or deliberately rendered inoperative, this fact is made known to the operator by the "Safeguards Equipment Locked Open" light indicator in the control room and the tagging of the AFW pump control switch. No modifications have been proposed which could result in the interaction of the AFWS safety function with control functions.

Capability to ascertain the performance of the AFWS at the Indian Point Unit 2 (IP-2) facility is provided by flow indication (one flow indicating channel per steam generator) and steam generator level indication (one wide range and three narrow range level channels per steam generator) in the control room. All four AFWS flow channels are powered from a separate independent Class 1E 120 VAC Instrumentation and Control Bus (Instrument Buses 21, 22, 23, and 24). The licensee has stated that the flow channels are seismically and environmentally qualified, and satisfy the single failure criterion.

The AFWS flow channels are tested once per 18 months (refueling intervals) by injecting simulated flow signals into the flow transmitters and verifying flow indication at the particular indicator being tested. These flow channels have no control functions (remote and local indication only).

The IP-2 steam generator level instrumentation consists of three safety grade narrow range level channels (used to initiate protective functions) and one non-safety grade wide range level channel per steam generator. The wide range level channels are all powered from the same bus (Instrument Bus 23) and control room indication for these wide range channels is provided to the operator by two dual pen recorders (two wide range level channels per recorder). An evaluation of the Indian Point Unit 2 steam generator level instrumentation will be provided at a later date, following the issuance of Regulatory Guide 1.97 Revision 2 (R.G. 1.97 - "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident") as a requirement for operating reactors.

The licensee has proposed tehcnical specification changes to require testing of one logic channel of the low-low level AFWS automatic logic on a monthly basis. Coupled with other requirements already in the technical specifications, the staff concurs that there will be adequate testing of the AFW automatic actuation logic. Therefore the proposed technical specification satisfy our concerns in this area.

The environmental qualification of safety related systems including AFWS circuits and components is being reviewed as part of the review of licensee responses to "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," issued to the licensee in NRR letter dated March 5, 1980. A separate Safety Evaluation will be issued at a later date to cover this review.

Based on our review of the AFWS automatic initiation system at Indian Point Unit 2, we conclude that the initiation signals, logic, and associated circuitry comply with the long term safety grade requirements of NUREG-0578, Section 2.1.7.a and the subsequent clarification issued by the staff, and therefore, are 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 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 an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, 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: August 30, 1982

Principal Contributors:
B. Lefave
R. Kendall
J. Thoma

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-247

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 79 to Facility Operating License No. DPR-26, issued to the Consolidated Edison Company of New York, Inc. (the licensee), which revised Technical Specifications for operation of the Indian Point Nuclear Generating Unit No. 2 (the facility) located in Buchanan, Westchester County, New York. The amendment is effective 21 days from the date of issuance.

The amendment revises their Technical Specifications to require certain condensate valves to be open when the plant is above 350°F and provide for testing of the steam generator low level AFWS automatic actuation logic.

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 August 11, 1980, (2) Amendment No. 79 to License No. DPR-26, 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 30th day of August, 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

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