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Docket No. 50-220

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Mr. Donald P. Dise
Vice President - Engineering
Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

Dear Mr. Dise:

In response to your requests for license amendment dated May 19 and May 25, 1977, the Commission has issued the enclosed Amendment No. 26 to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station Unit No. 1.

This amendment incorporates provisions into the facility Technical Specifications which establish limiting conditions for operation and surveillance requirements for drywell to suppression chamber differential pressure control and suppression pool water level.

These requirements provide assurance that facility operation will be in accordance with the assumptions utilized in your facility's plant-unique analysis which was performed in conjunction with the Mark I Containment Short Term Program evaluation.

The enclosed license amendment reflects those changes to your original request for license amendment which have been agreed to in discussions with your staff. These changes have been made to provide consistent requirements for all Mark I containment facilities.

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

Sincerely,

Original signed by

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

7902260148

Enclosures:
1. Amendment No. 26 to DPR-63
2. Safety Evaluation
3. Notice

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DATE	1/26/79	1/26/79	1/ /79	1/29/79	1/26/79	

Mr. Donald P. Dise

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Oswego, New York 13126

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

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 26
License No. DPR-63

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

   A. The applications for amendment by Niagara Mohawk Power Corporation (the licensee) dated May 19 and May 25, 1977, comply 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 applications, 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-63 is hereby amended to read as follows:

   (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

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 29, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 26

FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

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

Pages

iii
129
130
131
132
133
188
190
236
237

Add pages:

164a
164b
164c
164d
232b
232c

LIMITING CONDITION FOR OPERATION

3.3.2 PRESSURE SUPPRESSION SYSTEM PRESSURE AND SUPPRESSION CHAMBER WATER TEMPERATURE AND LEVEL

Applicability:

Applies to the interrelated parameters of pressure suppression system pressure and suppression chamber water temperature and level.

Objective:

To assure that the peak suppression chamber pressure does not exceed design values in the event of a loss-of-coolant accident.

Specification:

a. The downcomers in the suppression chamber shall have a minimum submergence of three feet and a maximum submergence of four and one half feet whenever the reactor coolant system temperature is above 215F.

b. During normal power operation, the combination of primary containment pressure and suppression chamber water temperature shall be within the shaded area of

SURVEILLANCE REQUIREMENT

4.3.2 PRESSURE SUPPRESSION SYSTEM PRESSURE AND SUPPRESSION CHAMBER WATER TEMPERATURE AND LEVEL

Applicability:

Applies to the periodic testing of the pressure suppression system pressure and suppression chamber water temperature and level.

Objective:

To assure that the pressure suppression system pressure and suppression chamber water temperature and level are within required limits.

Specification:

a. At least once per day the suppression chamber water level and temperature and pressure suppression system pressure shall be checked.

b. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

c. Whenever heat from relief valve operation is being added to the suppression pool the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(1) Figure 3.3.2a when downcomer submergence is $\geq$ 4 feet, or (2) Figure 3.3.2b when downcomer submergence is $\geq$ 3 feet. If these temperatures are exceeded, pool cooling shall be initiated immediately.	d. Whenever operation of a relief valve is indicated and the suppression pool temperature reaches 160F or above while the reactor primary coolant system pressure is greater than 200 psig, an external visual examination of the suppression chamber shall be made before resuming normal power operation.
c. If Specifications a and b above are not met within 24 hours, the reactor shall be shutdown using normal shutdown procedures.	
d. During testing of relief valves which add heat to the torus pool, the water temperature shall not exceed 10F above the normal power operation limit specific in b above. In connection with such testing, the pool temperature must be reduced within 24 hours to below the normal power operation limit specified in b above.	
e. The reactor shall be scrammed from any operating condition when the suppression pool temperature reaches 110F. Operation shall not be resumed until the pool temperature is reduced to below the normal power operation limit specified in b above.	
f. During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the pool temperature reaches 120F.	

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Figure 3.3.2 a
ALLOWABLE PRESSURE SUPPRESSION SYSTEM
4 FOOT DOWNCOMER SUBMERGENCE

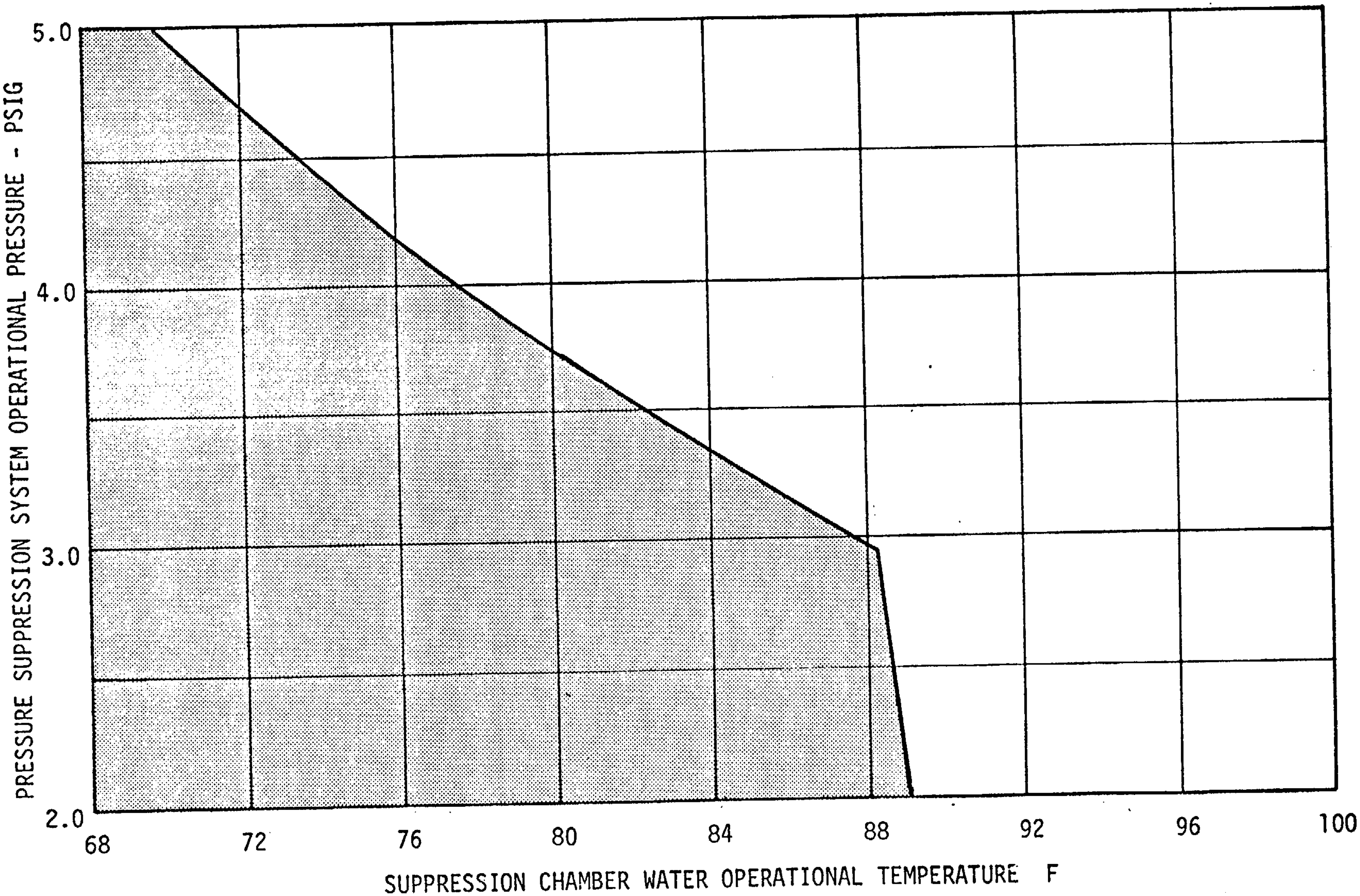
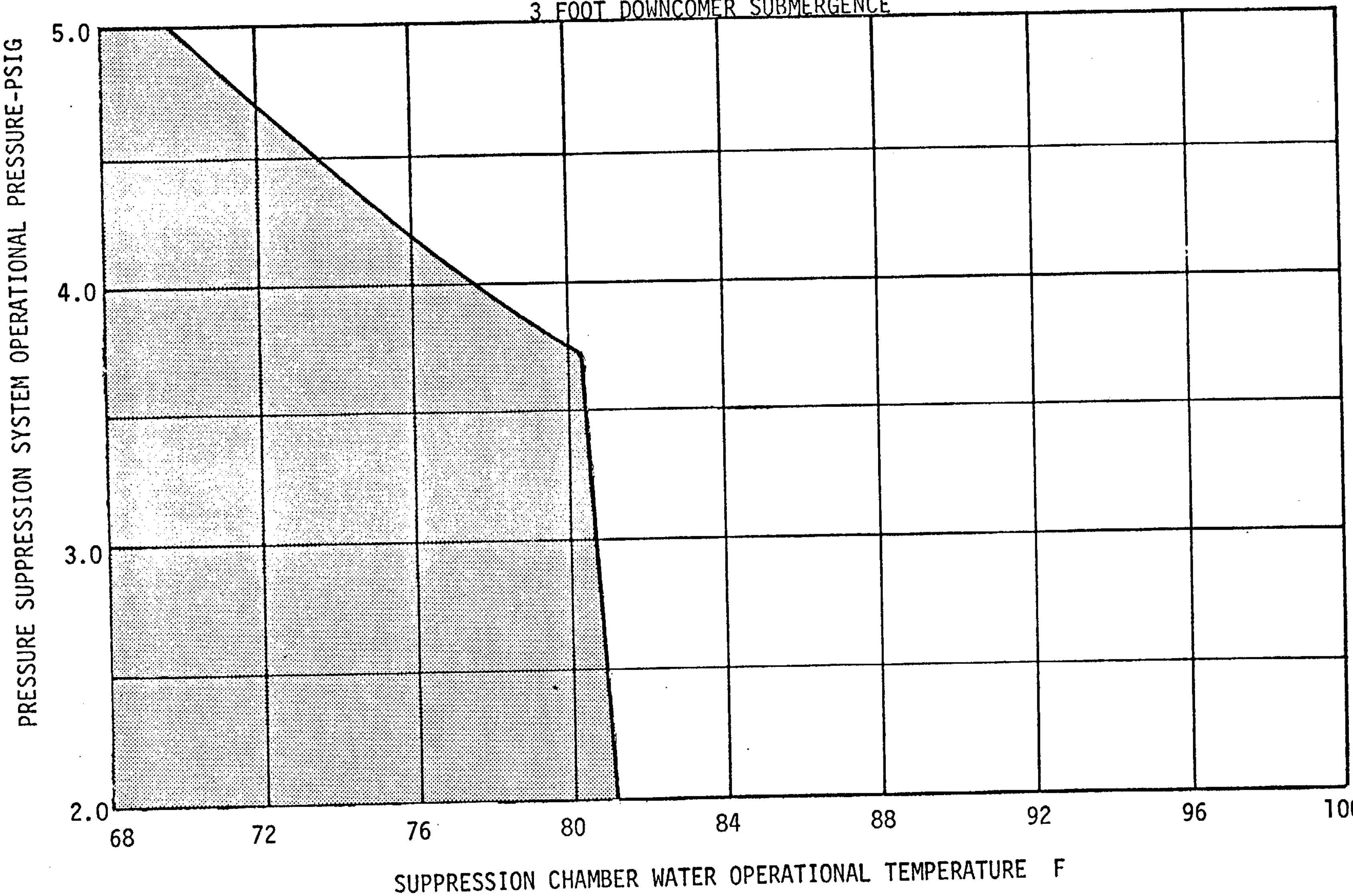


Figure 3.3.2 b
ALLOWABLE PRESSURE SUPPRESSION SYSTEM
3 FOOT DOWNCOMER SUBMERGENCE



LIMITING CONDITIONS FOR OPERATION

3.3.8 Drywell-Suppression Chamber Differential Pressure

Applicability:

Applies to the operational status of drywell-suppression chamber differential pressure system.

Objective:

To assure that the pressure suppression system will remain functional during a design basis loss-of-coolant accident.

Specification:

a. Differential pressure between the drywell and suppression chamber shall be maintained at or above levels according to Figure 3.3.8 except as specified in (1) and (2) below:

   (1) The differential pressure shall be established within 24 hours of achieving operating pressure and temperature. The differential pressure may be reduced to less than that specified by Figure 3.3.8 24 hours prior to a scheduled shutdown.

   (2) This differential may be decreased to less than that of Figure 3.3.8 for a maximum of four (4) hours during required operability testing of the drywell-pressure suppression chamber vacuum breakers.

SURVEILLANCE REQUIREMENTS

4.3.8 Drywell-Suppression Chamber Differential Pressure

Applicability:

Applies to the periodic testing requirements for the drywell-suppression chamber differential pressure system.

Objective:

To verify the operability of the drywell-suppression chamber differential system.

Specification:

a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.

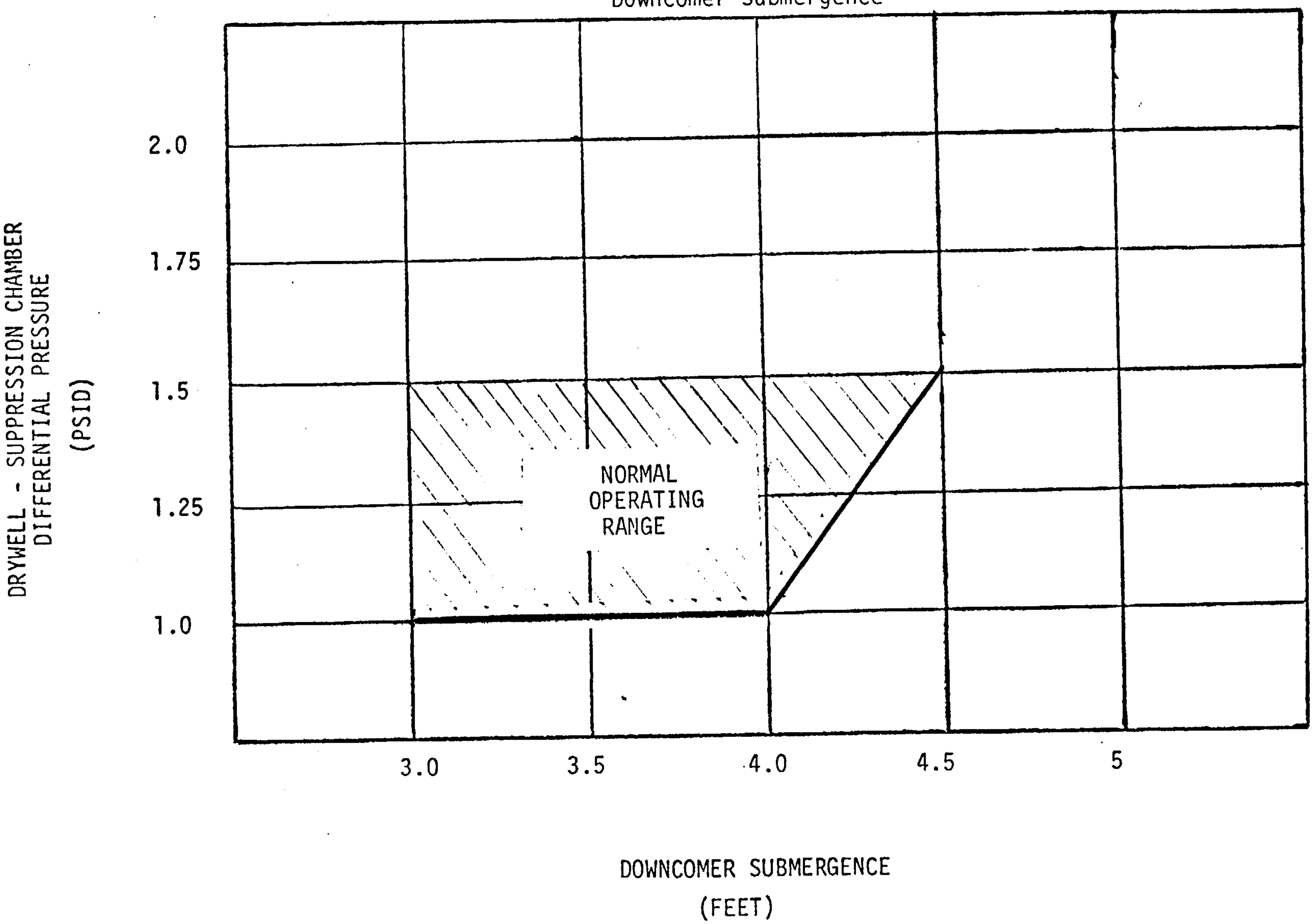
LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
b. If the differential pressure of Specification 3.3.8.a cannot be maintained and the differential pressure cannot be restored within the subsequent six (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in a Hot Shutdown condition in six (6) hours and a Cold Shutdown condition within the following 18 hours.	

FIGURE 3.3.8

Drywell-Suppression Chamber Differential Pressure

vs.

Downcomer Submergence



In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure in accordance with Figure 3.3.8 and suppression chamber water level corresponding to a downcomer submergence range of 3.0 to 4.5 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

3.6.2 PROTECTIVE INSTRUMENTATION

Applicability:

Applies to the operability of the plant instrumentation that performs a safety function.

Objective:

To assure the operability of the instrumentation required for safe operation.

Specification:

a. The set points, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Tables 3.6.2a to 3.6.21.

If the requirements of a table are not met, the actions listed below for the respective type of instrumentation shall be taken.

(1) Instrumentation that initiates scram-control rods shall be inserted.

4.6.2 PROTECTIVE INSTRUMENTATION

Applicability:

Applies to the surveillance of the instrumentation that performs a safety function.

Objective:

To verify the operability of protective instrumentation.

Specification:

a. Sensors and instrument channels shall be checked, tested and calibrated at least as frequently as listed in Tables 4.6.2a to 4.6.21.

(8) Off-Gas and Vacuum Pump Isolation - The respective system shall be isolated or the instrument channel shall be considered inoperable and Specification 3.6.1 shall be applied.

(9) Diesel Generator Initiation - The diesel generator shall be considered inoperable and Specification 3.6.3 shall be applied.

(10) Emergency Ventilation Initiation - The emergency ventilation system shall be considered inoperable and Specification 3.4.4 shall be applied.

(11) High Pressure Coolant Injection Initiation - The high pressure coolant injection system shall be considered inoperable and Specification 3.1.8.c shall be applied.

(12) Primary Containment Monitoring - The primary containment monitoring instrumentation shall be considered inoperable and Specification 3.3.8 shall be applied.

b. During operation with a Maximum Total Peaking Factor (MTPF) greater than the design value, either:

(1) The APRM scram and rod block settings shall be reduced to the values given by the equations in Specification 2.1.2.a; or

(2) The power distribution shall be changed such that the MTPF no longer exceeds the design value.

Table 3.6.21

PRIMARY CONTAINMENT MONITORING

Limiting Condition For Operation

	Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels Per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
					Shutdown	Refuel	Startup	Run
(1)	Drywell-Suppression Chamber Differential Pressure	2	1	Figure 3.3.8			X	X
(2)	Suppression Chamber Water Level	2	1	Specification 3.3.2			X	X

Table 4.6.21

PRIMARY CONTAINMENT MONITORING

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)	Drywell-Suppression Chamber Differential Pressure	once/day	N/A	Once Every Six Months
(2)	Suppression Chamber Water Level	once/day	N/A	Once Every Six Months

The set points on the generator load rejection and turbine stop valve closure scram trips are set to anticipate and minimize the consequences of turbine trip with failure of the turbine bypass system as described in the bases for Specification 2.1.2. Since the severity of the transients is dependent on the reactor operating power level, bypassing of the scrams below the specified power level is permissible.

The primary containment monitoring system is provided to alert the operator of conditions which could reduce safety margins during a postulated Loss of Coolant Accident. Appropriate operator corrective action is described in Specification 3.3.8, should Limiting Conditions for Operation be exceeded. This monitoring instrumentation does not automatically initiate engineered safeguards systems.

Although the operator will set the setpoints at the values indicated in Tables 3.6.2.a-1, the actual values of the various set points can differ appreciably from the value the operator is attempting to set. The deviations include inherent instrument error, operator setting error and drift of the set point. These errors are compensated for in the transient analyses by conservatism in the controlling parameter assumptions as discussed in the bases for Specification 2.1.2. The deviations associated with the set points for the safety systems used to mitigate accidents have negligible effect on the initiation of these systems. These safety systems have initiation times which are orders of magnitude greater than the difference in time between reaching the nominal set point and the worst set point due to error. The maximum allowable set point deviations are listed below:

Neutron Flux
 APRM, +2.7% of rated neutron flux
 IRM, +2.5% of rated neutron flux

Recirculation Flow, ± 1% of rated recirculation flow

Reactor Pressure, ±15.8 psig

Containment Pressure, ±0.053 psig

Reactor Water Level, ±2.6 inches of water

Main Steam Line Isolation Valve Position, ±2.5% of stem position

Scram Discharge Volume, + 0 and - 1 gallon

Condenser Low Vacuum, ±0.5 inches of mercury

High Flow-Main Steam Line, ±1 psid

High Flow-Emergency Cooling Line, ±1 psid

High Area Temperature-Main Steam Line, ±10F

High Area Temperature-Clean-up and Shutdown, ±6F

High Radiation-Main Steam Line, +100% and -50% of set point value

High Radiation-Emergency Cooling System Vent, +100% and -50% of set point

High Radiation-Reactor Building Vent, +100% and -50% of set point

High Radiation-Refueling Platform, +100% and -50% of set point

High Radiation-Offgas Line, ±50% of set point, (Appendix D)*

Drywell-Suppression Chamber Differential Pressure, ±0.1 psid

Suppression Chamber Water Level, ±1.8 inches

The test intervals for the trip systems result to calculated failure probabilities $\leq 10^{-4}$ which corresponds to the proposed IEEE Criteria For System Failure Probability. (IEEE SG-3, Information Docket #1 - Protection System Reliability, April 24, 1968).

The test intervals for the trip systems result in calculated failure probabilities ranging from 6.7×10^{-7} to 1.76×10^{-10} (Fifth Supplement, p. 115).* The more frequent sensor checks result in even less probability that the particular system will fail. Because of local high radiation, testing instrumentation in the area of the main steam line isolation valves can only be done during periods of Station shutdown. These functions include high area temperature isolation, high radiation isolation and isolation valve position scram.

Testing of the scram associated with the shutdown position of the mode switch can be done only during periods of Station shutdown since it always involves a scram.

*FSAR

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 26 TO LICENSE NO. DPR-63

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION UNIT NO. 1

DOCKET NO. 50-220

I. INTRODUCTION

In conjunction with the Short Term Program (STP) evaluation of Boiling Water Reactor facilities with the Mark I containment system, Niagra Mohawk Power Corporation (the licensee) submitted a Plant Unique Analysis (PUA) for the Nine Mile Point Nuclear Station Unit No. 1. This analysis was performed to confirm the structural and functional capability of the containment suppression chamber and attached piping, to withstand newly-identified suppression pool hydrodynamic loading conditions which had not been explicitly considered in the original design analysis for the plant. As part of the STP evaluation, specific loading conditions were developed for each Mark I facility, to account for the change in the magnitude of the loads due to plant-specific variations from the reference plant design for which the basic loading conditions were developed.

The results of the NRC staff's review of the hydrodynamic load definition techniques and the Mark I containment plant unique analyses are described in the "Mark I Containment Short Term Program Safety Evaluation Report," NUREG-0408, December 1977. As discussed in this report, the NRC staff has concluded that each Mark I containment system would maintain its integrity and functional capability in the unlikely event of a design basis loss-of-coolant accident (LOCA) and, therefore, that licensed Mark I BWR facilities can continue to operate safely, without undue risk to the health and safety of the public, during an interim period of approximately two years, while a methodical, comprehensive Long Term Program is conducted.

As discussed in Section III.C of NUREG-0408, of all of the plant parameters that were considered in the development of the hydrodynamic loads for the STP, only two parameters are expected to vary during normal plant operation; these are (1) the drywell-wetwell differential pressure; and (2) the suppression chamber (torus) water level. Subsequent to the submittal of the PUA, the licensee was requested to submit proposed Technical Specifications which assure that the allowable range

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of these two parameters during facility operation would be in accordance with the values utilized in the PUA.

The licensee has been operating this facility with differential pressure control to enhance the safety margins of the containment structure since early 1976. This evaluation provides a more detailed basis for establishing the allowable range of drywell-wetwell differential pressure and torus water level, in order to quantify containment safety margins. This amendment incorporates these parameters into the Technical Specifications with the associated limiting conditions for operation and surveillance requirements.

By letters dated May 19 and May 25, 1977, the licensee proposed changes to the facility Technical Specifications to incorporate limiting conditions for operation and surveillance requirements for differential pressure control and torus water level. Our evaluation of these proposed changes follows.

II. EVALUATION

The licensee has proposed certain Technical Specification requirements for the purpose of assuring that the normal plant operating conditions are within the envelope of conditions considered in their PUA. These Technical Specification changes establish (1) limiting condition for operation (LCOs) for drywell to torus differential pressure and torus water level, and (2) associated surveillance requirements. All other initial conditions utilized in the PUA are either presently included in the Technical Specifications or are configurational conditions which have been confirmed by the licensee and will not change during normal operation.

Differential pressure between the drywell and the suppression chamber will result in leakage of the drywell atmosphere to the lower pressure regions of the reactor building and to the torus airspace. This leakage from the drywell will cause a slow decay in the differential pressure. Therefore, surveillance requirements for the differential pressure have been included in the Technical Specifications. Surveillance frequency of once per operating shift for the differential pressure was selected on the basis of previous operating experience.

The torus water level is not expected to vary significantly during normal operation, unless certain systems connected to the suppression pools are activated. The torus water level would normally be monitored whenever such systems are in use. Therefore, we find that inclusion of periodic torus water level surveillance requirements in the Technical Specifications is not required.

We have reviewed the differential pressure and torus water level monitoring instrumentation systems proposed by the licensee with regard to the number of available channels and the instrumentation accuracy. This type of instrumentation is typically calibrated at six-month intervals. To assure proper operation during such intervals, two monitoring channels for both differential pressure and torus water level have been provided, such that a comparison of the readings will indicate when one of the channels is inoperative or drifting. The errors in the instrumentation are sufficiently small relative to the magnitude of the measurement (i.e., a maximum differential pressure measurement error of 0.1 psid in a measurement of 1.0 to 2.0 psid and a maximum torus water level measurement error of 10% of the difference between the maximum and minimum torus water level) that they may be neglected, based on the expected load variation with differential pressure and torus water level.

There are certain periods during normal plant operations when the differential pressure control cannot be maintained. Therefore, provisions have been included in the Technical Specifications to relax the differential pressure/control requirements during specified periods. The justification for relaxing the differential pressure control during these specific periods and the basis for selecting the duration of the periods are discussed in detail below.

A. <u>Startup and Shutdown</u>

During plant startup and shutdown, the drywell atmosphere undergoes significant barometric changes due to the variation in heat loads from the primary and auxiliary systems. In addition, it is during these periods that the drywell is being either inerted with nitrogen gas or deinerted. In order to keep the periods during which the differential pressure control is not fully effective as short as reasonable, we have limited the relaxation of the differential pressure control requirements for the startup and shutdown periods to 24 hours following startup and 24 hours prior to a shutdown. This time period was selected on a basis similar to that for the inerting requirements, already existing in the Technical Specifications. The postulated design basis accident for the containment assumes that the primary system is at operating pressure and temperature. During the startup and shutdown transients, the primary system is at operating pressure and temperature for only a part of the transient, during which the differential pressure is being established. These time periods have been shown by previous operating experience to be adequate with respect to the startup and shutdown transients, and at the same time sufficiently small in comparison to the duration of the average power run. Since the principal accident event to which differential pressure control is important to assure containment integrity (i.e., with a factor of safety of two) is a large break LOCA, we have considered whether there is a significantly greater probability of a large break LOCA during the startup and shutdown transients. We have concluded that there is not. Further, the operation of the plant systems is monitored more closely than normal during these periods and a finite magnitude of differential pressure will be available during the majority of these periods to mitigate the potential consequences of an accident.

B. Testing and Maintenance

During normal operation, there are a number of tests which are required to be conducted to demonstrate the continued functional performance of engineered safety features. The testing of certain systems will require, or result in, a reduction in the drywell-torus differential pressure. The operability testing of the drywell-torus vacuum breakers requires the removal of the differential pressure to permit the vacuum breakers to open. For the testing of high-energy systems (e.g. high pressure coolant injection pumps) during normal operation, the discharge flow is routed to the suppression pool. This energy deposition will raise the temperature of the suppression pool, resulting in an increase in torus pressure and a reduction in the differential pressure.

Functional performance testing of engineered safety features is necessary to assure proper maintenance of these systems throughout the life of the plant. Some of these tests (i.e., pump operability and drywell-wetwell vacuum breakers) may require or result in a reduction in the differential pressure. We estimate that not more than four tests will be required each month which will result in a reduction in differential pressure. In order to keep the periods during which the differential pressure control is not fully effective as short as is reasonable, we have permitted a relaxation of differential pressure control in order to conduct these tests, limited to a period of up to four hours. Again, we have carefully considered whether the probability of a large LOCA is significantly greater during these testing periods than that during normal operation. We conclude that it is not. Moreover, only the test of the drywell-wetwell vacuum breakers requires complete removal of the differential pressure.

Provisions have also been included in the Technical Specifications for performing maintenance activities on the differential pressure control system and for resolving operational difficulties which may result in an inadvertent reduction in the differential pressure for a short period of time. In certain circumstances, corrective action can be taken without having to attain a cold shutdown condition. To avoid repeated and unnecessary partial cooldown cycles, a restoration period has been incorporated into the action requirements of the LCO for differential pressure control; i.e., in the event that the differential pressure cannot be restored in six hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours. The six hour restoration period was selected on the basis that it represents an adequate minimum period of time during which any short-term malfunctions could be corrected, coupled with the minimum period of time required to conduct a controlled shutdown. The allowable time to conduct a controlled shutdown has been minimized, because the containment transients response is more a function of the primary system pressure than the reactor power level. On this basis, we find the proposed restoration period and action requirement acceptable.

We conclude that the limits imposed on the periods of time during which operation is permitted without the differential pressure control fully effective provides adequate assurance of overall containment integrity, and the periods of time differential pressure control is completely removed are acceptably small.

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 Section 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.

CONCLUSIONS

The proposed Technical Specifications will provide the necessary assurance that the plant's operating conditions remain within the envelope of the conditions assumed in the Plant Unique Analysis (PUA) performed in conjunction with the Mark I Containment Short Term Program. The PUA supplements the facility's Final Safety Analysis Report (FSAR) in that it demonstrates the plant's capability to withstand the suppression pool hydrodynamic loads which were not explicitly considered in the FSAR. We therefore conclude that the proposed changes to the Technical Specifications are acceptable.

We further conclude, 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: January 29, 1979

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-220

NIAGARA MOHAWK POWER CORPORATION

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-63, issued to Niagara Mohawk Power Corporation (the licensee), which revised the Technical Specifications for operation of the Nine Mile Point Nuclear Station Unit No. 1 (the facility), located in Oswego County, New York. The amendment is effective as of the date of its issuance.

The amendment revised the Technical Specifications to incorporate requirements for establishing and maintaining the drywell to suppression chamber differential pressure and suppression chamber water level, to maintain the margins of safety established in the Commission staff's "Mark I Containment Short Term Program Safety Evaluation," NUREG-0408. Operation in accordance with the conditions specified in NUREG-0408 has been previously authorized in 43 FR 13110 on March 29, 1978.

The applications for the amendment comply 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 Section 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) applications for amendment dated May 19 and 25, 1977, (2) Amendment No. 26 to License No. DPR-63, 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. 20555, and at the Oswego County Library, 46 E. Bridge Street, Oswego, New York 13126. 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 29 day of January 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors