March 20, 1995

Mr. B. Ralph Sylvia
Executive Vice President, Nuclear
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
P.O. Box 63
Lycoming, NY 13093

SUBJECT: ISSUANCE OF AMENDMENT FOR NINE MILE POINT NUCLEAR STATION UNIT NO. 1 (TAC NO. M89786)

Dear Mr. Sylvia:

The Commission has issued the enclosed Amendment No. 154 to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station Unit No. 1 (NMP-1). The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated June 30, 1994, as supplemented March 7, 1995.

The amendment revises Technical Specification (TS) 3.2.7.1 to add 8 check valves to Table 3.2.7.1. These valves were installed to add additional protection of the low pressure Core Spray system from the high pressure Reactor Coolant system. Including the valves in the TSs will assure that the proper surveillance testing is done to maintain a high reliability for the valves to protect the Core Spray system.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Original signed by

Gordon E. Edison, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-220

Enclosures: 1. Amendment No. 154 to DPR-63
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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cc w/encls: See next page

B. Ralph Sylvia
Niagara Mohawk Power Corporation

Nine Mile Point Nuclear Station
Unit No. 1

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DATED: March 20, 1995

AMENDMENT NO. 154 TO FACILITY OPERATING LICENSE NO. DPR-63-NINE MILE POINT UNIT NO. 1

Docket File
PUBLIC
PDI-1 Reading
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C. Vogan
G. Edison
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D. Hagan, T-4 A43
G. Hill (2), T-5 C3
C. Grimes, 11/E/22
R. Goel
ACRS (4)
OPA
OC/LFDCB
PD plant-specific file
C. Cowgill, Region I

cc: Plant Service list

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 154
License No. DPR-63

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

   A. The application for amendment by Niagara Mohawk Power Corporation (the licensee) dated June 30, 1994, as supplemented March 7, 1995, 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-63 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. 154, 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 to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

LB Marsh

Ledyard B. Marsh, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 20, 1995

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 154 TO FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

Revise Appendix A as follows:

Remove Page	Insert Page
117	117

TABLE 3.2.7.1

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

	System	Valve No.	Maximum[a] Allowable Leakage
1.	Core Spray System	40-03	≤5.0 gpm
		40-13	≤5.0 gpm
2.	Condensate Supply to Core Spray (Keep Fill System)	40-20	≤5.0 gpm
		40-21	≤5.0 gpm
		40-22	≤5.0 gpm
		40-23	≤5.0 gpm
3.	Core Spray Supply to Shutdown Cooling (Waterseal)	38-165	≤0.375 gpm
		38-166	≤0.375 gpm
		38-167	≤0.375 gpm
		38-168	≤0.375 gpm
		38-169	≤0.375 gpm
		38-170	≤0.375 gpm
		38-171	≤0.375 gpm
		38-172	≤0.375 gpm

Footnote:

(a) 1. Leakage rates shall be limited to 0.5 gpm per nominal inch of valve diameter up to a maximum of 5 gpm.

2. Test differential pressure shall not be less than 150 psid.

3. The observed leakage at test differential pressure shall be adjusted to the functional maximum pressure differential.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 154 TO FACILITY OPERATING LICENSE NO. DPR-63

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION UNIT NO. 1

DOCKET NO. 50-220

1.0 INTRODUCTION

In a submittal dated June 30, 1994, and supplemented March 7, 1995, Niagara Mohawk, the licensee for Nine Mile Point Unit 1 (NMP1) proposed to revise Technical Specification (TS) Table 3.2.7.1, "Primary Coolant System Pressure Isolation Valves," to add eight shutdown cooling pressure isolation check valves 38-165, 166, 167, 168, 169, 170, 171, and 172. These valves represent the high pressure/low pressure interface between the high pressure Reactor Coolant System and low pressure Core Spray System. This will allow the use of Core Spray System as a seal water system for Shutdown Cooling System containment isolation valves 38-01, 02, 12, and 13. The proposed changes also provide testing of these check valves in a manner similar to valves already contained in Table 3.2.7.1. The March 7, 1995, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Shutdown Cooling System does not perform any Loss-of-Coolant Accident (LOCA) function except for the containment isolation valves. The licensee indicated that the original NMP1 design basis considered the Shutdown Cooling System to be an extension of containment under accident conditions, and therefore the suction and discharge valves did not require testing in accordance with 10 CFR Part 50, Appendix J. In a Safety Evaluation dated May 6, 1988, the NRC staff determined that these valves are containment isolation valves and should be included in the Appendix J program for Type C local leak rate testing. In order to allow the replacement of the existing valves with valves capable of being Type C air tested, the licensee requested a schedular exemption from Appendix J due to time required for procurement and installation of testable valves and as low as is reasonable achievable (ALARA) concerns associated with draining and decontamination of the reactor vessel. A schedular exemption was granted which allowed the valves to be replaced during the 1995 refueling outage.

During a meeting with the NRC staff on February 22, 1994, the licensee proposed that rather than installing the previously planned modification, they would satisfy the requirements of Appendix J by installing a water seal pressurizing system between the inboard and outboard isolation valves. Section III.C.3 of 10 CFR Part 50, Appendix J, requires a continuous water seal for at least 30 days at a pressure of 1.10 Pa for valves utilizing a water seal. The licensee stated that the proposed modification will meet the requirements of Section III.C.3(a)(b) of Appendix J except when the unit is being cooled from approximately 300 °F to less than 215 °F during which time isolation would be provided by only a single valve. The licensee indicated that utilization of the proposed water seal would reduce radiological exposure by approximately 50 manrem, avoid a 14 day extension of refueling outage 13, save approximately 3 million dollars in direct cost required to install new valves and that there would be no overall safety advantage for installing the new isolation valves. The above does not include the replacement power costs which are estimated at 200,000 dollars per day and risks to plant personnel and equipment from rigging etc. associated with the replacement of four large containment isolation valves within the tight confines of the primary containment. In a letter dated March 21, 1994, the NRC staff indicated that if the licensee decided to pursue the proposed change for the shutdown cooling system rather than install the committed additional valves, it should submit a license amendment describing and justifying the proposed change.

2.0 EVALUATION

The licensee stated that the Core Spray System will provide the pressurized seal water from either core spray loop 11 or 12. The water seal will be applied to the interface between the inboard and outboard shutdown cooling isolation valves. The check valves are being added to provide the high pressure/low pressure interface and will also prevent seal water from one loop entering the other. An adequate water supply for 30 days for the Core Spray System will be available from the torus, since leakage through the valves would be back to the containment or to the closed loop Shutdown Cooling System. Additionally, make-up to the torus is provided from the Containment Spray Raw Water System. The seal water supply lines have been sized to provide a flow of 20 gpm at a pressure of 38.5 psig (1.1 Pa), based on a minimum pressure of approximately 150 psig at the discharge of the core spray topping pump. This corresponds to the theoretical minimum topping pump discharge pressure assumed in the Appendix K analysis at the run out flow of approximately 4540 gpm. The licensee indicated that in order to evaluate the effect of the reduction in core spray flow, it was assumed that a gross failure to seat occurred in one of the shutdown cooling isolation valves and resulted in a back pressure of zero psig. This would cause a maximum flow diversion of 35 gpm. The Appendix K analysis assumes approximately 10% degradation in flow (approximately 450 gpm) due to pump degradation, and therefore, the loss of 35 gpm is insignificant and has no adverse effect on the analysis.

The Core Spray System would be required to run continuously for 30 days in order to meet the 30-day requirement of Section III.C.3.(b) of 10 CFR Part 50, Appendix J. In the event that the accident requires either partial or no core spray injection (i.e., small break LOCA), the core spray pumps would typically be secured and seal water would not be available. The licensee indicated that in order to meet the 30-day requirement, throttling and extended recirculation modes of operation for the Core Spray System are being added to maintain an operable water seal system. The Core Spray System throttling capability is provided by modification of the test return valve logic and safety related qualification of the test return valves to allow remote manual operation of the valves with the core spray injection valves. Extended recirculation would require the test return valves to open which are both powered from power board 167, representing a potential single failure which could cause a loss of the water seal. If the small break is 1700 gpm or greater then the operator can throttle the injection valves and maintain total pump flow greater than 2200 gpm long term minimum flow requirement (1700 gpm break flow plus 500 gpm through the minimum flow line). The failure of power board 167 does not affect the availability of the minimum flow line or impact the ability of the operator to throttle core spray inboard injection valves. Therefore, no cycling of a core spray pump is required under these conditions. Core spray pump cycling would be required if the small break is less than 1700 gpm and the test return valves can not be opened due to the failure of power board 167. In this limiting condition, the isolation valves water seal can be maintained by keeping the reactor level within the emergency operating procedure (EOP) range of 53 to 95 inches by throttling the injection valves and cycling of the core spray pump with the restriction not to operate on minimum flow less than 2200 gpm for longer than 15 minutes. The maximum number of pump starts would allow each pump motor to be idle for 80 minutes which satisfies the condition for which no adverse effect on motor life or operability exists. This condition is assumed to exist for approximately 24 hours. In this time period it is expected that an alternate feedwater lineup using high pressure coolant injection, containment spray raw water, or diesel driven fire pumps can be established to maintain level within the 53 to 95 inches operating band.

The licensee indicated that the loss of power board 167 is considered a very unlikely event since this power board is safety related, seismically supported, and can be powered off either of the two onsite emergency diesel generators. Upon loss of the diesel generator which is supplying power to power board 167, automatic transfer of power board 167 will occur to the operating diesel. The failure of this board will be readily detected as it supplies both control and power to the red and green indicating lights in the control room for valves in various systems such as Core Spray, Containment Spray, Emergency Condenser, and loss of control power would result in various alarms and annunciators in the control room for these systems.

The licensee also indicated that in the very unlikely scenario of a small break LOCA concurrent with a failure of power board 167, the Core Spray test return line valves can be manually operated as these valves are physically and radiologically accessible during a small break LOCA assuming the entire core remains covered at all times. The probability of a small break LOCA combined with the failure of power board 167 is calculated to be low, 2.8×10^{-7} per year. However, this will not result in a core damage event. If test return valves 40-05 and 40-06 can be manually operated then the water seal is maintained for 30 days and there will be no airborne leakage from the primary containment through the Shutdown Cooling System containment isolation valves.

Even if the seal water system were to fail (i.e., multiple single failures) then airborne leakage through these isolation valves would be limited since these valves are water tested to a Technical Specification acceptance criterion of 5 gpm per valve. This leakage would also be confined to the Shutdown Cooling System since this system is a closed loop system and has a design rating of 1250 psig. Also the Shutdown Cooling System is totally enclosed within secondary containment and is seismically supported. In the event of leakage from this system which becomes airborne, any radioactivity would be filtered, if needed, by the operation of the Emergency Ventilation System.

The licensee indicated that elimination of the single failure of power board 167 would require design, purchase and installation of a safety related power board. Safety related cabling and conduit would also have to be installed as well as modification work in the control room. Elimination of this single failure would cost approximately one million dollars. The NRC staff finds the deviation that the power board 167 is not single failure proof acceptable due to low probability of its failure, available indications in the control room, limited radiological consequences, and high cost of modifications.

The licensee indicated that during plant operation the Shutdown Cooling System isolation valves are normally closed and breakers for the valves 38-01, 02, and 13 will be racked out to prevent a spurious valve opening (single active failure) from defeating the water seal. Valve 38-12 is a check valve. The water seal is subject to a single active failure when the plant is in the process of cooldown and the breakers have been racked in such that the system can perform its function. This occurs when the reactor coolant temperature is less than 350°F and reactor pressure is approximately 120 psig. Should a LOCA occur at this time, failure of an isolating valve to close upon receipt of an initiating signal could cause a loss of water seal. Based on current operating experience, less than three startup/shutdown cycles are expected per year that results in an average of 10 hours per year when the system could be in this configuration. A probabilistic risk assessment was performed by the licensee to determine the probability of a LOCA occurring during the time when the Shutdown Cooling System is in operation. The calculated probability of a non-watersealed core damage event is 1.1×10^{-9} per year. The NRC staff finds the isolation provided by a single isolation valve during the above acceptable.

The proposed seal water system for Shutdown Cooling System containment isolation valves 38-01, 02, 12 and 13 will utilize the low pressure Core Spray System. In order to utilize this system, it is necessary to install check valves 38-165, 166, 167, 168, 169, 170, 171, and 172. The addition of these valves represents an additional high pressure/low pressure interface, thus creating the potential for an intersystem LOCA. Therefore, these valves are being added to TS Table 3.2.7.1 and will be tested in a manner similar to the valves already contained in that Table to ensure low probability of gross failure and so reduce the risk of an intersystem LOCA. The licensee stated that the portion of the seal water system from the interface with the

Shutdown Cooling System containment isolation valves up to and including the Check valves identified above is designed as safety related with a temperature and pressure rating equal to that of the Reactor Coolant System. This configuration is similar to that used in the Keep Fill System that was added to prevent water hammer in the Core Spray System (i.e., two check valves downstream of the motor operated core spray injection valves). The proposed change does not affect operation of either the Shutdown Cooling System or Core Spray System.

Based on the above evaluation, the NRC staff finds acceptable the proposed seal water system and changes to add Shutdown Cooling System check valves 38-165, 166, 167, 168, 169, 170, 171 and 172 to TS Table 3.2.7.1 to allow utilization of the Core Spray System as a seal water system for Shutdown Cooling System containment isolation valves 38-01, 02, 12 and 13. The addition of these pressure isolation valves, although a physical change, does not alter the initial conditions used for any design basis accident. Leak testing of the valves in accordance with TS 3.2.7.1 provides assurance that the Core Spray System will not be damaged by an overpressurization event which could lead to potential loss of integrity of the system.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (59 FR 39593). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: R. Goel

Date: March 20, 1995