January 25, 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. M89980)

Dear Mr. Sylvia:

The Commission has issued the enclosed Amendment No. 152 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 (TSs) in response to your application transmitted by letter dated July 21, 1994.

The amendment revises TSs 2.2.2, 3.2.8, 4.2.8, and the associated Bases to reduce the number of reactor head safety valves required operable from 16 valves to 9 valves. The setpoints of the valve groups are unchanged by this amendment. The amendment requires testing of the safety valves in accordance with the approved NMP-1 Inservice Test Program.

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

Donald S. Brinkman, 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. 152 to DPR-63
2. Safety Evaluation

cc w/encls: See next page

Distribution: See attached sheet

DOCUMENT NAME: G:\NMP1\NM189980.AMD *See previous concurrence

CP1

To receive a copy of this document, indicate in the box: "C" = Copy without enclosures "E" = Copy with enclosures "N" = No copy

OFFICE	LA:PDI-1	E	PM:PDI-1	E	BC:SRXB*		OGC*		D:PDI-1	N		
NAME	CVogan		DBrinkman:smm		RJones		SUttal		LMarsh			
DATE	~~12/ /94~~ 1/25/95		01/25/95		01/07/95		01/10/95		01/25/95			

OFFICIAL RECORD COPY

9501310152 950125
PDR ADOCK 05000220
P PDR

300034

DF01

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

January 25, 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. M89980)

Dear Mr. Sylvia:

The Commission has issued the enclosed Amendment No. 152 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 (TSs) in response to your application transmitted by letter dated July 21, 1994.

The amendment revises TSs 2.2.2, 3.2.8, 4.2.8, and the associated Bases to reduce the number of reactor head safety valves required operable from 16 valves to 9 valves. The setpoints of the valve groups are unchanged by this amendment. The amendment requires testing of the safety valves in accordance with the approved NMP-1 Inservice Test Program.

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,

Donald S. Brinkman

Donald S. Brinkman, 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. 152 to DPR-63
2. Safety Evaluation

cc w/encls: See next page

B. Ralph Sylvia
Niagara Mohawk Power Corporation

Nine Mile Point Nuclear Station
Unit No. 1

cc:

Mark J. Wetterhahn, Esquire
Winston & Strawn
1400 L Street, NW
Washington, DC 20005-3502

Supervisor
Town of Scriba
Route 8, Box 382
Oswego, NY 13126

Mr. Louis F. Storz
Vice President - Nuclear Generation
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
P.O. Box 63
Lycoming, NY 13093

Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 126
Lycoming, NY 13093

Gary D. Wilson, Esquire
Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, NY 13202

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Ms. Donna Ross
New York State Energy Office
2 Empire State Plaza
16th Floor
Albany, NY 12223

Mr. Richard B. Abbott
Unit 1 Plant Manager
Nine Mile Point Nuclear Station
P.O. Box 63
Lycoming, NY 13093

Mr. David K. Greene
Manager Licensing
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
P.O. Box 63
Lycoming, NY 13093

Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, NY 10271

Mr. Paul D. Eddy
State of New York
Department of Public Service
Power Division, System Operations
3 Empire State Plaza
Albany, NY 12223

Mr. Martin J. McCormick, Jr.
Vice President
Nuclear Safety Assessment
and Support
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
P.O. Box 63
Lycoming, NY 13093

DATED: January 25, 1995

AMENDMENT NO. 152 TO FACILITY OPERATING LICENSE NO. DPR-63-NINE MILE POINT UNIT NO. 1

Docket File
PUBLIC
PDI-1 Reading
S. Varga, 14/E/4
J. Zwolinski
L. Marsh
C. Vogan
D. Brinkman
OGC
D. Hagan, 3302 MNBB
G. Hill (2), P1-22
C. Grimes, 11/F/23
K. Kavanagh
ACRS (10)
OPA
OC/LFDCB
PD plant-specific file
C. Cowgill, Region I
R. Jones
J. Strosnider

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. 152
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 July 21, 1994, 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) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 152, 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

LBMarsh

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: January 25, 1995

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 152 TO FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

Revise Appendix A as follows:

Remove Pages	Insert Pages
23	23
25	25
118	118
119	119

SAFETY LIMIT

2.2.1 REACTOR COOLANT SYSTEM

Applicability:

Applies to the limit on reactor coolant system pressure.

Objective:

To define those values of process variables which shall assure the integrity of the reactor coolant system to prevent an uncontrolled release of radioactivity.

Specification:

The reactor vessel or reactor coolant system pressure shall not exceed 1375 psig at any time with fuel in the vessel.

LIMITING SAFETY SYSTEM SETTING

2.2.2 REACTOR COOLANT SYSTEM

a. The settings on the safety valves of the pressure vessel shall be as shown below. The allowable initial set point error on each setting will be ±1 percent.

Set Point (Psig)	Number of Safety Valves
1218	3
1227	2
1236	2
1245	1
1254	1
	9

b. The reactor high-pressure scram trip setting shall be ≤1080 psig.

c. The flow biased APRM scram trip settings shall be in accordance with Specification 2.1.2a.

a. The range of set points for a safety valve actuation is selected in accordance with code requirements. The safety valves are sized according to the code for a condition of main steam isolation valve closure while operating at 1850 Mwt, followed by a reactor scram on high neutron flux. Under these conditions, a total of nine (9) safety valves are required to limit reactor pressure below the safety limit of 1375 psig.

In addition to the safety valves, the solenoid-actuated relief valves are used to prevent safety valve lift during rapid reactor isolation at power coupled with failure of the bypass system. Any five of these valves opening at 1090 psig to 1100 psig will keep the maximum vessel pressure below the lowest safety valve setting, as demonstrated in Appendix E-I.3.11 (p. E-35)*. (The Technical Supplement to Petition to Increase Power Level, and letter from T. J. Brosnan, Niagara Mohawk Power Corporation, to Peter A. Morris, Division of Reactor Licensing, USAEC, dated February 28, 1972). Subsequently, six valves were provided due to the blowdown requirements, following a small line break.

b. The reactor high pressure scram setting is relied upon to terminate rapid pressure transients if other scrams, which would normally occur first, fail to function. As demonstrated in Appendix E-I of the FSAR and the Technical Supplement to Petition to Increase Power Level, Page II-12, the reactor high pressure scram is a backup to the neutron flux scram, generator load rejection scram, and main steam isolation-valve closure scram for various reactor isolation incidents. However, rapid isolation at lower power levels generally results in high pressure scram preceding other scrams because the transients are slower and those trips associated with the turbine-generator are bypassed.

The operator will set the trip setting at 1080 psig or lower. However, the actual set point can be as much as 15.8 psi above the 1080 psig indicated set point due to the deviations discussed above.

*FSAR

LIMITING CONDITION FOR OPERATION

3.2.8 PRESSURE RELIEF SYSTEMS-SAFETY VALVES

Applicability:

Applies to the operational status of the safety valves.

Objective:

To assure the capability of the safety valves to limit reactor overpressure below the safety limit in the event of rapid reactor isolation and failure of all pressure relieving devices.

Specification:

a. During power operating conditions and whenever the reactor coolant pressure is greater than 110 psig and temperature greater than saturation temperature all nine of the safety valves shall be operable.

b. If specification 3.2.8a is not met, the reactor coolant pressure and temperature shall be reduced to 110 psig or less and saturation temperature or less, respectively, within ten hours.

SURVEILLANCE REQUIREMENT

4.2.8 PRESSURE RELIEF SYSTEMS-SAFETY VALVES

Applicability:

Applies to the periodic testing requirements for the safety valves.

Objective:

To assure the capability of the safety valves to limit reactor overpressure to below the safety limit.

Specification:

At least once during each operating cycle, the number of safety valves as determined by the IST Program Plan shall be removed, tested for set point and partial lift, and then returned to operation or replaced.

BASES FOR 3.2.8 AND 4.2.8 PRESSURE RELIEF SYSTEM-SAFETY VALVES

The required number of operable safety valves is based on a condition of main steam isolation valve closure while operating at 1850 Mwt, followed by a reactor scram on high neutron flux. Operation of all 9 safety valves will limit reactor pressure below the safety limit of 1375 psig.

The safety valve testing and intervals between tests are based on manufacturer's recommendations and past experience with spring actuated safety valves.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 152 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

By letter dated July 21, 1994, Niagara Mohawk Power Corporation (the licensee or NMPC) submitted a request for changes to the Nine Mile Point Nuclear Station Unit No. 1 (NMP-1), Technical Specifications (TSs). The requested changes would eliminate seven reactor head safety valves from TS 2.2.2, "Reactor Coolant System," TS 3.2.8/4.2.8, "Pressure Relief Systems - Safety Valves," and the associated Bases for these sections. The current plant configuration includes 16 safety valves and 6 pressure relief valves. The original design for NMP-1 overpressure protection was based on American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section I, 1962 Edition (Ref. 2). ASME Code, Section I, deals specifically with power boilers and not nuclear reactors. (In 1962, ASME Code, Section III, which deals with nuclear reactors did not exist.) This edition was interpreted to require that the safety valve capacity be such that all potentially generated steam by the boiler (reactor vessel) be discharged without credit for fuel stoppage (reactor scram). The current versions of the ASME Code (Ref. 3), Sections I and III, allow credit for operating and/or safety controls in the boiler or nuclear reactor. Based on the current ASME Code and NUREG-0800, "Standard Review Plan" (Ref. 4), NMPC proposes to take credit for the high flux scram and plans to remove 7 out of 16 safety valves provided at NMP-1. NMPC proposed that the reduction of the number of safety valves would result in considerable reduction in man-rem exposure and savings due to reduced maintenance and surveillance testing.

The NMP-1 safety valves are grouped into 5 groups with each group of safety valves set to relieve at staggered groups of pressures as specified in TS 2.2.2. The proposed amendment would not change the specified relief set points for the 5 groups of valves; the proposed amendment would only decrease the number of valves in each group so that the total number of valves would be 9 rather than 16.

TS 4.2.8 would be revised to require periodic testing of the safety valves in accordance with the NMP-1 Inservice Testing Program rather than having TS 4.2.8 specify that at least 8 of the 16 valves be tested at least once per operating cycle.

2.0 EVALUATION

The overpressure protection system prevents overpressurization of the reactor coolant pressure boundary under the most severe transients and limits the reactor pressure during normal operational transients. Safety valves are required to be designed with sufficient capacity to limit the pressure to less than 110 percent of the reactor coolant pressure boundary design pressure of 1250 psig. Currently, overpressure protection is provided by 16 safety and 6 relief valves located on the reactor head and on the two main steam lines between the reactor vessel and the first isolation valve inside the drywell. The relief valves discharge to the suppression pool and the safety valves discharge to the drywell. The combined relief and safety valve capacity is approximately $13.5x10^6$ lb/hr and equivalent to 185 percent of the total steam flow. The probability of lifting the safety valves is low due to the turbine bypass capability of 40 percent and the 6 relief valves. The proposed combined relief and safety valve capacity after removing the 7 safety valves would be approximately $9x10^6$ lb/hr and equivalent to 124 percent of the total steam flow.

Since the only function of the safety valves at NMP-1 is to provide ASME Code overpressure protection, NMPC performed an overpressure protection analysis to verify that 9 safety valves are sufficient to meet the acceptance criteria for overpressure protection. The impact on Anticipated Transient Without Scram (ATWS) was also examined.

2.1 Overpressure Protection - Main Steam Line Isolation with High Neutron Flux Scram

The peak pressure with 9 safety valves was calculated using General Electric approved methodology documented in NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel (GESTAR)," February 1991. The analysis was performed assuming that: a) the reactor is in operation at the design rated power of 1850 MW_{th} and design pressure of 1250 psig, b) main steam line isolation occurs without a scram, c) all pressure relieving devices fail, and d) the reactor scrammed on high neutron flux (120% of rated flux). The high neutron flux signal is the second safety grade scram signal from the reactor protection system following main steam isolation valve (MSIV) closure. Normally a reactor scram would occur on MSIV closure of 10 percent. The analysis only took credit for the 9 safety valves. The results of the analysis demonstrated that the calculated peak pressure of the vessel remained below the 1375 psig limit (110 percent of the vessel design pressure). For the most severe transient, main steam line isolation with a high neutron flux scram at 120 percent, the calculated peak pressure is 1353 psig when only

9 safety valves are assumed to operate in the safety mode. Since the calculated peak pressure, 1353 psig, is within the acceptance criterion of 1375 psig, the overpressure protection analysis is acceptable.

2.2 Anticipated Transient Without Scram (ATWS) - Main Steam Line Isolation with High Neutron Flux Scram and Recirculation Pump Trip

Another plant upset which could be impacted by this change is the postulated ATWS event. While not part of the plant's original design basis, requirements were imposed by 10 CFR 50.62 to reduce the likelihood and consequences of an ATWS. The Code of Federal Regulations at 10 CFR 50.62 required that all boiling water reactors have an alternate rod injection (ARI) system, a standby liquid control system (SLCS), and automatic reactor recirculation pump trips. An analysis was performed assuming that: a) the reactor is in operation at the design rated power of 1850 MW_{th} and design pressure of 1250 psig, b) main steam line isolation occurs without a scram, and c) credit taken for recirculation pump trip and six relief valve actuations. Normally, the ARI logic at NMP-1 would cause the alternate rod injection valves to energize when either the reactor vessel high pressure trip setpoint or the low water level trip setpoint is reached. Once energized, insertion of the control rods begins within 15 seconds. For this transient, the peak calculated pressure was 1322 psig when only 9 safety valves are assumed instead of 16 safety valves and the ARI fails. This result is consistent with the GE ATWS Event Evaluations (NEDE-24222). Therefore, based on the analyses presented above, the staff concludes that operation with 9 safety valves instead 16 safety valves will not endanger the public health and safety.

2.3 Revision to Safety Valve Surveillance Requirement

The proposed change to TS 4.2.8 would require the safety valves to be tested in accordance with the NMP-1 Inservice Testing (IST) Program. The NMP-1 IST Program is based on the ASME Code, Section XI, 1983, including Summer Addenda which is in accordance with the requirements of 10 CFR 50.5a(f) and is, therefore, acceptable.

3.0 SUMMARY

NMPC performed an overpressure analysis consistent with Standard Review Plan 5.2.2 to support its proposal to remove 7 safety valves at NMP-1. The setpoints, as specified in TS 2.2.2, for the 5 groups of safety valves were not changed in this analysis. The results of the overpressure analysis were consistent with the NRC staff's acceptance criteria that pressure not exceed 110 percent of design. The licensee also discussed the impact on ATWS response, indicating no significant impact. Periodic surveillance testing of the safety valves will be performed in accordance with the NMP-1 IST Program. Therefore, NMPC's proposal to remove 7 safety valves from the current

16 safety valves is acceptable. The changes proposed in TS 2.2.2 and 3.2.8/4.2.8 reducing the number of safety valves and the bases are also acceptable.

4.0 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.

5.0 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 and changes a surveillance requirement. 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 45027). 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.

6.0 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.

7.0 REFERENCES

1. Letter from B. R. Sylvia (NMPC) to U.S. NRC, "Nine Mile Point Unit 1," dated July 21, 1994.

2. ASME Boiler and Pressure Vessel Code, Section I, "Safety Valves," American Society of Mechanical Engineers, 1962, pgs. 75 - 80.

3. ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

4. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," June 1987.

Principal Contributors: K. Kavanagh
D. Brinkman

Dated: January 25, 1995