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

July 26, 1993

Docket No. 50-220

Mr. B. Ralph Sylvia
Executive Vice President, Nuclear
Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

Dear Mr. Sylvia:

SUBJECT: ISSUANCE OF AMENDMENT FOR NINE MILE POINT NUCLEAR STATION UNIT NO. 1 (TAC NO. M85074)

The Commission has issued the enclosed Amendment No. 143 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 November 24, 1992, as supplemented June 30, 1993.

The amendment revises the TSs to utilize a revised correlation for the Flow Biased Scram and Average Power Range Monitors Rod Block functions. The revisions are an integral part of the process computer upgrade that was implemented during the recent refueling outage (Reload 13). The process computer upgrade utilizes the "3D Monicore" software which was supplied as part of the computer upgrade.

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

Enclosures:
1. Amendment No. 143 to DPR-63
2. Safety Evaluation

cc w/enclosures:
See next page

DF01
1/1

cc:

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

Supervisor
Town of Scriba
Route 8, Box 382
Oswego, New York 13126

Vice President - Nuclear Generation
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
Post Office Box 32
Lycoming, New York 13093

Resident Inspector
U.S. Nuclear Regulatory Commission
Post Office Box 126
Lycoming, New York 13093

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

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

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

Mr. Kim Dahlberg
Unit 1 Station Superintendent
Nine Mile Point Nuclear Station
Post Office Box 32
Lycoming, New York 13093

Mr. David K. Greene
Manager Licensing
Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

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

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

DATED: July 26, 1993

AMENDMENT NO. 143 TO FACILITY OPERATING LICENSE NO. DPR-63-NINE MILE POINT UNIT 1

Docket File
NRC & Local PDRs
PDI-1 Reading
S. Varga, 14/E/4
J. Calvo, 14/A/4
R. Capra
C. Vogan
D. Brinkman
OGC
D. Hagan, 3302 MNBB
G. Hill (2), P1-22
Wanda Jones, P-370
C. Grimes, 11/F/23
R. Jones, 8/E/23
A. Attard, 8/E/23
ACRS (10)
OPA
OC/LFDCB
PD plant-specific file
C. Cowgill, Region I

cc: Plant Service list

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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. 143
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 November 24, 1992, as supplemented June 30, 1993, 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. 143, 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

Robert A. Capra

Robert A. Capra, 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: July 26, 1993

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 143 TO FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

Revise Appendix A as follows:

Remove Pages	Insert Pages
2	2
9	9
10	10
11	11
12	12
14	14
18	18
19	19
20	20
21	21
22	22
23	23
26	26
65	65
66	66
67	67
195	195
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197	197
200	200
203	203
227	227
366	366

f. Linear Heat Generation Rate (LHGR)

The heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the fuel length.

g. Average Planar Linear Heat Generation Rate (APLHGR)

The Average Planar Linear Heat Generation Rate (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the heat generation rate per unit length of fuel rod for all fuel rods in the specified bundle at the specified height, divided by the number of fuel rods in the fuel bundle at that height.

h. Critical Power

That assembly power which causes some point in the assembly to experience transition boiling.

i. Critical Power Ratio (CPR)

The ratio of critical power to the bundle power at the reactor condition of interest.

j. Minimum Critical Power Ratio (MCPR)

The minimum in-core critical power ratio.

k. Fraction of Limiting Power Density (FLPD)

The linear heat generation rate (LHGR) existing at a given location divided by the specified LHGR limit for that bundle type.

l. Core Maximum Fraction of Limiting Power Density (CMFLPD)

The highest value of the fraction of limiting power density which exists in the core.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

2.1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variables associated with fuel thermal behavior.

Objective:

To establish limits on the important thermal-hydraulic variables to assure the integrity of the fuel cladding.

Specification:

a. When the reactor pressure is greater than 800 psia and the core flow is greater than 10%, the existence of a Minimum Critical Power Ratio (MCPR) less than the Safety Limit Critical Power Ratio (SLCPR) (Reference 12) shall constitute violation of the fuel cladding integrity safety limit.

b. When the reactor pressure is less than or equal to 800 psia or core flow is less than 10% of rated, the core power shall not exceed 25% of rated thermal power.

2.1.2 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip settings on automatic protective devices related to variables on which the fuel loading safety limits have been placed.

Objective:

To provide automatic corrective action to prevent exceeding the fuel cladding safety limits.

Specification:

Fuel cladding limiting safety system settings shall be as follows:

a. The flow biased APRM scram and rod block trip setting shall be established according to the following relationships:

$S \leq (0.55W + 65\%)T$ with a maximum value of 120%

$S_{RB} \leq (0.55W + 55\%)T$ with a maximum value of 110%

WHERE:

S or S_{RB} = The respective scram or rod block setpoint

W = Loop Recirculation Flow as a percentage of the loop recirculation flow which produces a rated core flow of 67.5 MLB/HR

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

c. The neutron flux shall not exceed its scram setting for longer than 1.5 seconds as indicated by the process computer. When the process computer is out of service, a safety limit violation shall be assumed if the neutron flux exceeds the scram setting and control rod scram does not occur.

To ensure that the Safety Limit established in Specifications 2.1.1a and 2.1.1b is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.

d. Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be more than 6 feet, 3 inches (-10 inches indicator scale) below minimum normal water level (Elevation 302'9") except as specified in "e" below.

e. For the purpose of performing major maintenance (not to exceed 12 weeks in duration) on the reactor vessel; the reactor water level may be lowered 9' below the minimum normal water level (Elevation 302'9"). Whenever the reactor water level is to be lowered below the low-low-low level setpoint redundant instrumentation will be provided to monitor the reactor water level.

T = FRTP/CMFLPD (T is applied only if less than or equal to 1.0)
FRTP = Fraction of Rated Thermal Power where Rated Thermal Power equals 1850 MW
CMFLPD = Core Maximum Fraction of Limiting Power Density

With CMFLPD greater than the FRTP for a short period of time, rather than adjusting the APRM setpoints, the APRM gain may be adjusted so that APRM readings are greater than or equal to 100% times CMFLPD provided that the adjusted APRM reading does not exceed 100% of rated thermal power and a notice of adjustment is posted on the reactor control panel.

b. The IRM scram trip setting shall not exceed 12% of rated neutron flux.

c. The reactor high pressure scram trip setting shall be $\leq$ 1080 psig.

d. The reactor water low level scram trip setting shall be no lower than -12 inches (53 inches indicator scale) relative to the minimum normal water level (302'9").

e. The reactor water low-low level setting for core spray initiation shall be no less than -5 feet (5 inches indicator scale) relative to the minimum normal water level (Elevation 302'9").

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

Written procedures will be developed and followed whenever the reactor water level is lowered below the low-low level set point (5 feet below minimum normal water level). The procedures will define the valves that will be used to lower the vessel water level. All other valves that have the potential of lowering the vessel water level will be identified by valve number in the procedures and these valves will be red tagged to preclude their operation during the major maintenance with the water level below the low-low level set point.

In addition to the Facility Staff requirements given in Specification 6.2.2.b, there shall be another control room operator present in the control room with no other duties than to monitor the reactor vessel water level.

f. The reactor low pressure setting for main-steam-line isolation valve closure shall be ≥850 psig when the reactor mode switch is in the run position.

g. The main-steam-line isolation valve closure scram setting shall be ≤10 percent of valve closure (stem position) from full open.

h. The generator load rejection scram shall be initiated by the signal for turbine control valve fast closure due to a loss of oil pressure to the acceleration relay any time the turbine first stage steam pressure is above a value corresponding to 833 Mwt, i.e., 45 percent of 1850 Mwt.

i. The turbine stop valve closure scram shall be initiated at ≤10 percent of valve closure setting (Stem position) from full open whenever the turbine first stage steam pressure is above a value corresponding to 833 Mwt, i.e., 45 percent of 1850 Mwt.

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However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where similar fuel operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1400 psia during normal power operation (the limit of applicability of the boiling transition correlation), it would be assumed that the fuel cladding integrity safety limit has been violated.

In addition to the boiling transition limit SLCPR, operation is constrained to ensure that actual fuel operation is maintained within the assumptions of the fuel rod thermal-mechanical design and the safety analysis basis. At full power, this limit is the linear heat generation rate limit with overpower transients constrained by the unadjusted APRM scram and rod block. During steady-state operation at lower power levels, where the fraction of rated thermal power is less than the core maximum fraction of limiting power density, the APRM flow biased scram and rod block settings are adjusted by the equation in Specification 2.1.2a.

At pressure equal to or below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all core flows, this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and all flows will always be greater than 4.56 psi.

Analyses show that with a bundle flow of $28x10^3$ lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Therefore, due to the 4.56 psi driving head, the bundle flow will be greater than $28x10^3$ lb/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at $28x10^3$ lb/hr is approximately 3.35 MWt. With the design peaking factor, this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia or core flow less than 10% is conservative.

However, in response to expressed beliefs[7] that variation of APRM flux scram with recirculation flow is a prudent measure to assure safe plant operation during the design confirmation phase of plant operation, the scram setting will be varied with recirculation flow.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity safety limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity safety limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of FRTP and CMFLPD. The scram setting is adjusted in accordance with Specification 2.1.1a when the core maximum fraction of limiting power density exceeds the fraction of rated thermal power.

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at a constant recirculation flow rate, and thus to protect against the condition of a MCPR less than the SLCPR. This rod block trip setting, which is automatically varied with recirculation flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the safety limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which could occur during steady-state operation is at 110% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the core maximum fraction of limiting power density exceeds the fraction of rated thermal power, thus, preserving the APRM rod block safety margin.

b. Normal operation of the automatic recirculation pump control will be in excess of 30% rated flow; therefore, little operation below 30% flow is anticipated. For operation in the startup mode while the reactor is at low pressure, the IRM scram setting is 12% of rated neutron flux. Although the operator will set the IRM scram trip at 12% of rated neutron flux or less, the actual scram setting can be as much as 2.5% of rated neutron flux greater. This includes the margins discussed above. This provides adequate margin between the setpoint and the safety limit at 25% of rated power. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. There are a few possible sources of rapid reactivity input to the system in the low power flow condition. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit.

Procedural controls will assure that the IRM scram is maintained up to 20% flow. This is accomplished by keeping the reactor mode switch in the startup position until 20% flow is exceeded and the APRM's are on scale. Then the reactor mode switch may be switched to the run mode, thereby switching scram protection from the IRM to the APRM system.

In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to 1% of rated power, thus maintaining a limit above the SLCPR. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

c. As demonstrated in UFSAR Section XV-A and B, the reactor high pressure scram is a backup to the neutron flux scram, turbine stop valve closure 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.

d. A reactor water low level scram trip setting -12 inches (53 inches indicator scale) relative to the minimum normal water level (Elevation 302-9") will assure that power production will be terminated with adequate coolant remaining in the core. The analysis of the feedwater pump loss in UFSAR Section XV-B.3.13 has demonstrated that approximately 4 feet of water remains above the core following the low level scram.

   The operator will set the low level trip setting no lower than -12 inches relative to the lowest normal operating level. However, the actual set point can be as much as 2.6 inches lower due to the deviations discussed above.

e. A reactor water low-low level signal -5 feet (5 inches indicator scale) relative to the minimum normal water level (Elevation 302'9") will assure that core cooling will continue even if level is dropping. Core spray cooling will adequately cool the core, as discussed in LCO 3.1.4.

   The operator will set the low-low level core spray initiation point at no less than -5 feet (5 inches indicator scale) relative to the minimum normal water level (Elevation 302'9"). However, the actual set point can be as much as 2.6 inches lower due to the deviations discussed above.

f-g. The low pressure isolation of the main steam lines at 850 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line isolation on reactor low pressure and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at ≤10% valve closure, there is no increase in neutron flux and peak pressure if the vessel dome is limited to 1141 psig. (8, 9, 10).

The operator will set the pressure trip at greater than or equal to 850 psig and the isolation valve stem position scram setting at less than or equal to 10% of valve stem position from full open. However, the actual pressure set point can be as much as 15.8 psi lower than the indicated 850 psig and the valve position set point can be as much as 2.5% of stem position greater. These allowable deviations are due to instrument error, operator setting error and drift with time.

In addition to the above mentioned Limiting Safety System Setting, other reactor protection system devices (LCO 3.6.2) serve as a secondary backup to the Limiting Safety System Setting chosen. These are as follows:

High fission product activity released from the core is sensed in the main steam lines by the high radiation main steam line monitors. These monitors provide a backup scram signal and also close the main steam line isolation valves.

The scram dump volume high level scram trip assures that scram capability will not be impaired because of insufficient scram dump volume to accommodate the water discharged from the control rod drive hydraulic system as a result of a reactor scram (Section X-C.2.10)*.

h. The generator load rejection scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to the worst case transient of a load rejection and subsequent failure of the bypass. In fact, analysis [9,10] shows that heat flux does not increase from its initial value at all because of the fast action of the load rejection scram; thus, no significant change in MCPR occurs.

i. The turbine stop valve closure scram is provided for the same reasons as discussed in h above. With a scram setting of ≤10% valve closure, the resultant transients are nearly the same as for those described in h above; and, thus, adequate margin exists.

*UFSAR

REFERENCES FOR BASES 2.1.1 AND 2.1.2 FUEL CLADDING

(1) General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application, NEDO-10958 and NEDE-10958.

(2) Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10801, February 1973.

(3) UFSAR Section XV-A and B.

(4) UFSAR Section XV-A and B.

(5) UFSAR Section XV-A and B.

(6) UFSAR Section XV-A and B.

(7) Letters, Peter A. Morris, Director of Reactor Licensing, USAEC, to John E. Logan, Vice-President, Jersey Central Power and Light Company, dated November 22, 1967 and January 9, 1968.

(8) UFSAR Section XV-A and B.

(9) Letter, T. J. Brosnan, Niagara Mohawk Power Corporation, to Peter A. Morris, Division of Reactor Licensing, USAEC, dated February 28, 1972.

(10) Letter, Philip D. Raymond, Niagara Mohawk Power Corporation, to A. Giambusso, USAEC, dated October 15, 1973.

(11) Nine Mile Point Nuclear Power Station Unit 1 Load Line Limit Analysis, NEDO 24012, May, 1977.

(12) Licensing Topical Report "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, latest approved revision.

(13) Nine Mile Point Nuclear Power Station Unit 1, Extended Load Line Limit Analysis, License Amendment Submittal (Cycle 6), NEDO-24185, April 1979.

(14) General Electric SIL 299 "High Drywell Temperature Effect on Reactor Vessel Water Level Instrumentation."

(15) Letter (and attachments) from C. Thomas (NRC) to J. Charnley (GE) dated May 28, 1985, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-B, Amendment 10."

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

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.

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	4
1227	3
1236	3
1245	3
1254	3
	16

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.

c. As shown in Sections XV-B.3.1 and 3.5*, rapid Station transients due to isolation valve or turbine trip valve closures result in coincident high-flux and high-pressure transients. Therefore, the APRM trip, although primarily intended for core protection, also serves as backup protection for pressure transients.

Although the operator will set the scram setting at less than or equal to that required by Specification 2.1.2a, the actual neutron flux setting can be as much as 2.7 percent of rated neutron flux above the specified value. This includes the errors discussed above. The flow bias could vary as much as one percent of rated recirculation flow above or below the indicated point.

In addition to the above-mentioned Limiting Safety System Setting, other reactor protection system devices (LCO 3.6.2) serve as secondary backup to the Limiting Safety System Setting chosen. These are as follows:

The primary containment high-pressure scram serves as backup to high reactor pressure scram in the event of lifting of the safety valves. As discussed in Section VIII-A.2.1*, a pressure in excess of 3.5 psig due to steam leakage or blowdown to the drywell will trip a scram well before the core is uncovered.

A low condenser vacuum situation will result in loss of the main reactor heat sink, causing an increase in reactor pressure. The scram feature provided, therefore, anticipates the reactor high-pressure scram. A loss of main condenser vacuum is analyzed in Section XV-B.3.1.8*.

The scram dump volume high-level scram trip assures that scram capability will not be impaired because of insufficient scram dump volume to accommodate the water discharge from the control-rod-drive hydraulic system as a result of a reactor scram (Section X-C.2.10)*.

In the event of main-steam-line isolation valve closure, reactor pressure will increase. A reactor scram is, therefore, provided on main-steam-line isolation valve position and anticipates the high reactor pressure scram trip.

*UFSAR

LIMITING CONDITION FOR OPERATION

3.1.7 FUEL RODS

Applicability:

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective:

The objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specification:

a. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of axial location and average planar exposure shall not exceed the limiting value provided in the Core Operating Limits Report. When hand calculations are required, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) shown in the Core Operating Limits Report. If at any time during power operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded at any node in the core, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR at all nodes in the core is

SURVEILLANCE REQUIREMENT

4.1.7 FUEL RODS

Applicability:

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective:

The objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specification:

a. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of axial location and average planar exposure shall be determined daily during reactor operation at ≥ 25 percent rated thermal power.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
not returned to within the prescribed limits within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until APLHGR at all nodes is within the prescribed limits.	
b. Linear Heat Generation Rate (LHGR) During power operation, the Linear Heat Generation Rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the limiting value specified in the Core Operating Limits Report. If at any time during power operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded at any location, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR at all locations is not returned to within the prescribed limits within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until LHGR at all locations is within the prescribed limits.	b. Linear Heat Generation Rate (LHGR) The LHGR as a function of core height shall be checked daily during reactor operation at ≥25% rated thermal power.
c. Minimum Critical Power Ratio (MCPR) During power operation, the MCPR for all fuel at rated power and flow shall be within the limit provided in the Core Operating Limits Report. If at any time during power operation it is determined by normal surveillance that the above limit is no longer met, action shall be initiated within 15 minutes to restore operation to within	c. Minimum Critical Power Ratio (MCPR) MCPR shall be determined daily during reactor power operation at >25% rated thermal power.

LIMITING CONDITION FOR OPERATION

the prescribed limit. If all the operating MCPRs are not returned to within the prescribed limit within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until MCPR is within the prescribed limit.For core flows other than rated, the MCPR limit shall be the limit identified above times K_f where K_f is provided in the Core Operating Limits Report.

d. Power Flow Relationship During Operation

This power/flow relationship shall not exceed the limiting values shown in the Core Operating Limits Report.

If at any time during power operation it is determined by normal surveillance that the limiting value for the power/flow relationship is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the power/flow relationship is not returned to within the prescribed limits within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until the power/flow relationship is within the prescribed limits.

e. Partial Loop Operation

During power operation, partial loop operation is permitted provided the following conditions are met.

SURVEILLANCE REQUIREMENT

d. Power Flow Relationship

Compliance with the power flow relationship in Section 3.1.7.d shall be determined daily during reactor operation.

e. Partial Loop Operation

Under partial loop operation, surveillance requirements 4.1.7, a, b, c and d above are applicable.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(2) Primary Coolant and Containment Isolation - Isolation valves shall be closed or the valves shall be considered inoperable and Specifications 3.2.7 and 3.3.4 shall be applied.	b. Each trip system shall be tested each time the respective instrument channel is tested.
(3) Emergency Cooling Initiation or Isolation - The emergency cooling system shall be considered inoperable and Specification 3.1.3 shall be applied.	
(4) Core Spray Initiation - The core spray system shall be considered inoperable and Specification 3.1.4 shall be applied.	
(5) Containment Spray Initiation - The containment spray system shall be considered inoperable and Specification 3.3.7 shall be applied.	
(6) Auto Depressurization Initiation - The auto depressurization system shall be considered inoperable and Specification 3.1.5 shall be applied.	
(7) Control Rod Withdrawal Block - No control rods shall be withdrawn.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(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) Control Room Ventilation - The control room ventilation system shall be considered inoperable and Specification 3.4.5 shall be applied.	

LIMITING CONDITION FOR OPERATION

b. During operation with the Core Maximum Fraction of Limiting Power Density (CMFLPD) greater than the Fraction of Rated Thermal Power (FRTP), either:

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

(2) The APRM gain shall be adjusted in accordance with Specification 2.1.2a; or

(3) The power distribution shall be changed such that the CMFLPD no longer exceeds FRTP.

SURVEILLANCE REQUIREMENT

c. During reactor power operation at ≥ 25 percent rated thermal power, the Core Maximum Fraction of Limiting Power Density (CMFLPD) shall be checked daily and the flow-referenced APRM scram and rod block signals shall be adjusted, if necessary, as specified by Specification 2.1.2a.

TABLE 3.6.2a (cont'd)

INSTRUMENTATION THAT INITIATES SCRAM

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
	(ii) Inoperative	2	3(d)(o)	---		x	x	
	(b) APRM							
	(i) Upscale	2	3(e)(o)	Specification 2.1.2a		x	x	x
	(ii) Inoperative	2	3(e)(o)	---		x	x	x
	(iii) Downscale	2	3(e)(o)	≥ 5 percent of full scale		(g)	(g)	(g)
(10)	Turbine Stop Valve Closure	2	4(o)	≤ 10% valve closure				(i)
(11)	Generator Load Rejection	2	2(o)	(j)				(i)

NOTES FOR TABLES 3.6.2a and 4.6.2a

(a) May be bypassed when necessary for containment inerting.

(b) May be bypassed in the refuel and shutdown positions of the reactor mode switch with a keylock switch.

(c) May be bypassed in the refuel and startup positions of the reactor mode switch when reactor pressure is less than 600 psi, or for the purpose of performing reactor coolant system pressure testing and/or control rod scram time testing with the reactor mode switch in the refuel position.

(d) No more than one of the four IRM inputs to each trip system shall be bypassed.

(e) No more than two C or D level LPRM inputs to an APRM shall be bypassed and only four LPRM inputs to an APRM shall be bypassed in order for the APRM to be considered operable. No more than one of the four APRM inputs to each trip system shall be bypassed provided that the APRM in the other instrument channel in the same core quadrant is not bypassed. A Traversing In-Core Probe (TIP) chamber may be used as a substitute APRM input if the TIP is positioned in close proximity to the failed LPRM it is replacing.

(f) Calibrate prior to startup and normal shutdown and thereafter check once per shift and test once per week until no longer required.

(g) IRM's are bypassed when APRM's are onscale. APRM downscale is bypassed when IRM's are onscale.

(h) Each of the four isolation valves has two limit switches. Each limit switch provides input to one of two instrument channels in a single trip system.

(i) May be bypassed when reactor power level is below 45%.

(j) Trip upon loss of oil pressure to the acceleration relay.

(k) May be bypassed when placing the reactor mode switch in the SHUTDOWN position and all control rods are fully inserted.

(l) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2a, the primary sensor will be calibrated and tested once per operating cycle.

(m) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during reactor operation when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 2.1.2a shall not be included in determining the absolute difference.

TABLE 3.6.2g (cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD WITHDRAWAL BLOCK

Limiting Condition for Operation

	Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System (i)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
					Shutdown	Refuel	Startup	Run
	c. Downscale	2	3(b)	≥ 5 percent of full scale for each scale		x	x	
	d. Upscale	2	3(b)	≤ 88 percent of full scale for each scale		x	x	
(3)	APRM							
	a. Inoperative	2(h)	3(c)	---		x	x	x
	b. Upscale (Biased by Recirculation Flow)	2(h)	3(c)	Specification 2.1.2a (h)		x	x	x
	c. Downscale	2(h)	3(c)	≥ 2 percent of full scale		(d)	(d)	x

Changes to the Offsite Dose Calculation Manual (ODCM): Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the Offsite Dose Calculation Manual to be changed, together with appropriate analyses or evaluations justifying the change(s);

b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

c. Documentation of the fact that the change has been reviewed and found acceptable.

f. CORE OPERATING LIMITS REPORT

1. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle for the following:

1) The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.1.7.a and 3.1.7.e.

2) The K_f core flow adjustment factor for Specification 3.1.7.c.

3) The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.1.7.c and 3.1.7.e.

4) The LINEAR HEAT GENERATION RATE for Specification 3.1.7.b.

5) The Power/Flow relationship for Specification 3.1.7.d and e.

and shall be documented in the CORE OPERATING LIMITS REPORT.

2. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 143 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 November 24, 1992, as supplemented June 30, 1993, Niagara Mohawk Power Corporation (the licensee) submitted a request for changes to the Nine Mile Point Nuclear Station Unit No. 1 (NMP1), Technical Specifications (TSs). The requested changes would revise the TSs to utilize a revised correlation for the Flow Biased Scram and Average Power Range Monitors (APRMs) Rod Block functions. The revisions are an integral part of the process computer upgrade that was implemented during the 1993 refueling outage (Reload 13). The process computer upgrade utilizes the "3D Monicore" software which was supplied as part of the computer upgrade. The June 30, 1993, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The proposed changes are necessitated by new fuel bundle designs and new core monitoring techniques introduced by General Electric. New fuel designs with different active fuel lengths and variable enrichments require monitoring of parameters different from those required with simpler fuels. Consequently, the licensee is proposing revisions to the Limiting Safety System Settings associated with the APRMs as well as to the definitions of the parameters used in the determination of the Limiting Safety System Settings. In addition, the proposed amendment would replace the visual representation of the APRM Limiting Safety System Settings with the actual correlation used to determine the settings. The licensee also proposed to enhance the calculation of the average planar linear heat generation rate (APLHGR) by including axial location as an input parameter to the calculation.

The proposed changes are consistent with guidance contained in the approved NRC Safety Evaluation, Amendment No. 19 to GESTAR II. The proposed TSs will result in the same operating power distribution limits and safety margins as the current TSs. In addition, the proposed TS changes will reduce TS complexity and will preclude the need for inclusion of numerous lattice specific MAPLHGR curves.

2.1 Replacement of Figure 2.1.1

APRM Flow Biased Scram and Rod Block Safety Limits are provided in TS Figure 2.1.1. Analysis and data provided by the licensee shows that during operation at 100 percent rated thermal power, and 100 percent core flow with a 120 percent scram trip setting, none of the operational transients analyzed violate the fuel safety limit and that there is a substantial margin from fuel damage. The scram and rod block settings are varied with circulation flow to account for lower thermal margins at lower flows. Furthermore, the licensee showed in their submittal that to ensure that actual setpoints are not exceeded, the scram and rod block settings are clamped at their 100 percent core flow values.

To take advantage of current advances in human factor engineering, the licensee has replaced TS Figure 2.1.1 with the actual correlation used to determine the settings. References to TS Figure 2.1.1 are revised to refer to TS 2.1.2a, the location of the correlation. However, TS Figure 2.1.1 is retained in the NMP1 Updated Final Safety Analysis Report as Figure VIII-14, "Trip Logic for APRM Scram and Rod Block."

The licensee's APRM flow biased scram and rod block settings are adjusted ("setdown") to assure that the peak linear heat generation rate (LHGR) during a postulated transient event, is not increased for any combination of the maximum total peaking factor and reactor thermal power. The setdown factor adjusts the scram and rod block settings whenever that factor is less than one. Consequently, whenever calculated peaking factors exceed design values, the scram setpoints are reduced to maintain an equivalent margin of safety.

The NMP1 plant process computer assumes an active fuel length of 144 inches for calculation of the maximum total peaking factor. The utilization of advanced fuel design, with different fuel lengths, has necessitated the calculation of the scram and rod block setdown requirements in a different way. In essence, the existing setdown factor calculation is replaced with an equivalent factor "T" to accommodate the different fuel length designs.

T = FRTP/CMFLPD

Where:

FRTP = Fraction of Rated Thermal Power, and
CMFLPD = Core Maximum Fraction of Limiting Power Density

The correlation for the APRM scram and rod block Limiting Safety System Setting is being revised (as part of this submittal) to include this new setdown factor. The revisions to implement this change have been included in the proposed amendment, along with the revised Bases sections relating to the scram and rod block Limiting Safety System Settings.

2.2 Thermal Performance Parameters

In this submittal, the licensee has requested that axial location be added as parameter to use when determining the average planar linear heat generation rate (APLHGR) for a fuel bundle.

Until now, NMP1 obtained APLHGR by averaging the linear heat generation rate (LHGR) for each of the fuel rods of a particular fuel bundle type and selecting a limiting value maximum average planar linear heat generation ratio (MAPLHGR) as a function of fuel burnup. Newer GE fuel bundles have MAPLHGRs that vary axially depending upon the specific combination of enriched uranium and gadolinia that comprises a fuel bundle lattice cross section at a particular axial node. Each particularly enriched bundle is called a lattice type, having a MAPLHGR that varies axially and with fuel burnup. TS 3.1.7a has been revised to include the axial location as a parameter to be used when determining APLHGR and the definition of the APLHGR has been added to Section 1.0, "Definitions."

To minimize the probability of errors during hand calculations of the APLHGR, the licensee has provided additional directions, such that the allowed APLHGR for any bundle will not be allowed to exceed the limiting lattice APLHGR for that bundle. These changes are consistent with the guidance provided in the NRC Safety Evaluation of Amendment No. 19 to GESTAR II.

The NRC staff has reviewed the licensee's proposed TS changes necessitated by new fuel bundle designs and new core monitoring techniques introduced by GE. Based on this review, we conclude that the proposed changes satisfy staff positions and requirements in these areas and are therefore acceptable.

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 (57 FR 61116). 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:
A. Attard

Date: July 26, 1993

Docket No. 50-220 July 26, 1993

Mr. B. Ralph Sylvia
Executive Vice President, Nuclear
Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

Dear Mr. Sylvia:

SUBJECT: ISSUANCE OF AMENDMENT FOR NINE MILE POINT NUCLEAR STATION UNIT NO. 1 (TAC NO. M85074)

The Commission has issued the enclosed Amendment No. 143 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 November 24, 1992, as supplemented June 30, 1993.

The amendment revises the TSs to utilize a revised correlation for the Flow Biased Scram and Average Power Range Monitors Rod Block functions. The revisions are an integral part of the process computer upgrade that was implemented during the recent refueling outage (Reload 13). The process computer upgrade utilizes the "3D Monicore" software which was supplied as part of the computer upgrade.

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

Enclosures:
1. Amendment No. 143to DPR-63
2. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	PDI-1:LA	PDI-1:PM	OGC	PDI-1:D	
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DATE	7/7/93	7/8/93	7/12/93	7/26/93	/ /

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