Mr. John H. Mueller
Chief Nuclear Officer
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
Operations Building, Second Floor
P.O. Box 63
Lycoming, NY 13093

September 21, 1999

SUBJECT: ISSUANCE OF AMENDMENT FOR NINE MILE POINT NUCLEAR STATION, UNIT NO. 1 (TAC NO. MA4218)

Dear Mr. Mueller:

The Commission has issued the enclosed Amendment No.168 to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station, Unit No. 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated November 16, 1998, as supplemented by letter dated June 21, 1999.

This amendment changes the TSs regarding the setpoints for the average power range monitor flow biased scram to limit the magnitude of reactor power oscillations during a reactor trip, and along with changes to the control rod block settings, allow for operation in the Extended Load Line Limit Analysis region of the power/flow operating curve.

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

Sincerely,

Original signed by:

Darl S. Hood, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-220

Enclosures: 1. Amendment No.168 to DRP-63
2. Safety Evaluation

cc w/encls: See next page

19

DOCUMENT NAME: G:\PDI-1\NMP1\AMDA4218.WPD

DF01

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

OFFICE	PM:PDI	E	LA:PDI		SC:PDI		OGC			
NAME	DHood:lcc DSH		SLittle		SBajwa		R Bachmann			
DATE	9 / 9 /99		9 / 8 /99		9 / 20 /99		9 / 14 /99		/ /99	

Official Record Copy

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

September 21, 1999

Mr. John H. Mueller
Chief Nuclear Officer
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
Operations Building, Second Floor
P.O. Box 63
Lycoming, NY 13093

SUBJECT: ISSUANCE OF AMENDMENT FOR NINE MILE POINT NUCLEAR STATION, UNIT NO. 1 (TAC NO. MA4218)

Dear Mr. Mueller:

The Commission has issued the enclosed Amendment No. 168 to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station, Unit No. 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated November 16, 1998, as supplemented by letter dated June 21, 1999.

This amendment changes the TSs regarding the setpoints for the average power range monitor flow biased scram to limit the magnitude of reactor power oscillations during a reactor trip, and along with changes to the control rod block settings, allow for operation in the Extended Load Line Limit Analysis region of the power/flow operating curve.

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

Sincerely,

Darl S. Hood

Darl S. Hood, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-220

Enclosures: 1. Amendment No. 168 to DRP-63
2. Safety Evaluation

cc w/encls: See next page

Nine Mile Point Nuclear Station
Unit No. 1

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

Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 126
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. F. William Valentino, President
New York State Energy, Research,
and Development Authority
Corporate Plaza West
286 Washington Avenue Extension
Albany, NY 12203-6399

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

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

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

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

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 168
License No. DRP-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 16, 1998, as supplemented by letter dated June 21, 1999, 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 1;

   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. DRP-63 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No.168are hereby incorporated into this license. Niagara Mohawk Power Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

SBajwa

S. Singh Bajwa, Chief
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 21, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 168

TO FACILITY OPERATING LICENSE NO. DRP-63

DOCKET NO. 50-220

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

Remove	Insert
9	9
10	10
11	11
12	12
14	14
17	17
18	18
19	19
22	22
26	26
203	203
251	251
---	251a (new)
367	367

SAFETY LIMIT

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.

LIMITING SAFETY SYSTEM SETTING

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 settings shall be established according to the following relationships:

The minimum of:

For $W \geq 0\%$:

$S \leq (0.55W + 67\%)T$ with a maximum value of 122%

$S_{RB} \leq (0.55W + 62\%)T$ with a maximum value of 117%

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.

AND:

For $18\% \leq W \leq 40\%$:

$S \leq (1.287W + 20.83\%)$
$S_{RB} \leq (1.287W + 13.54\%)$

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

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.

SAFETY LIMIT

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.

LIMITING SAFETY SYSTEM SETTING

b. The IRM scram trip setting shall not exceed 12% of rated neutron flux for IRM range 9 or lower.

   The IRM scram trip setting shall not exceed 38.4% of rated neutron flux for IRM range 10.

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").

f. The reactor low pressure setting for main-steam-line isolation valve closure shall be $\geq$850 psig when the reactor mode switch is in the run position or the IRMs are on range 10.

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

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
	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.

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 equations 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 28×10^3 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 28×10^3 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 28×10^3 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.

The abnormal operational transients applicable to operation of the plant have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition of 1850 MWt. The analyses were based upon plant operation in accordance with the operating map given in Reference 11. In addition, 1850 MWt is the licensed maximum power level, and represents the maximum steady-state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in Reference 2.

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 20% insertion. By the time the rods are 60% inserted, approximately four dollars of negative reactivity have been inserted which strongly turns the transient, and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

a. The Average Power Range Monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated thermal power. Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses [5, 6, 8, 9, 10, 11, 13, 18] demonstrate that with a 122% scram trip setting, none of the abnormal operational transients analyzed violate the fuel safety limit and there is a substantial margin from fuel damage.

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.

Also, a scram setting has been established to preclude thermal-hydraulic instabilities which could compromise fuel safety limits. Specifically, the scram setting will limit the oscillation magnitude at reactor trip, thereby limiting the associated CPR change, and in conjunction with MCPR operating limits, assure compliance with the MCPR safety limit.

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 117% 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 (<800 psia), the IRM range 9 high flux (16, 17) scram setting is calibrated to correspond to 12% of rated neutron flux. The IRM range 9, 12% of rated neutron flux calibration is on a nominal basis, which provides adequate margin between the setpoint and the safety limit at 25% of rated power. The margin is also adequate to accommodate anticipated maneuvers associated with 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 for low flow condition. This is accomplished by keeping the IRMs on range 9 until 20% flow is exceeded and reactor pressure is >850 psig and that control rods shall not be withdrawn if recirculation flow is less than 30%. If the APRMs are onscale, then the reactor mode switch may be placed in run, thereby switching scram protection from the IRM to the APRM system. If the APRMs are not onscale, then operation with the mode switch in startup (including normal startup mode steam chest warming and bypass valve operation) may continue using IRM range 10, provided that the main turbine generator is not placed in operation.

To continue operation with the mode switch in startup beyond 12% of rated neutron flux, the IRMs must be transferred into range 10. The Reactor Protection System is designed such that reactor pressure must be above 850 psig to successfully transfer the IRMs into range 10, thus assuring added protection for the fuel cladding safety limit. The RPS design will cause the low reactor pressure main-steam-line isolation to be unbypassed when one IRM in trip system 11 and one IRM in trip system 12 are placed in range 10. Procedural controls assure that IRM range 9 is maintained on all IRM channels up to 850 psig reactor pressure. The IRM scram remains active until the mode switch is placed in the RUN position at which time the scram function is transferred to APRMs.

The adequacy of the IRM scram in range 10 (approximately 38.4% of rated neutron flux) was determined by comparing the scram level on the IRM range 10 to the minimum APRM scram level for transient protection. The APRM scram level for transient protection is defined by the Section 2.1.2a equation for $W \geq 0\%$. This equation results in a minimum APRM scram of 67% of rated power at zero recirculation flow. Therefore, startup mode transients (i.e., those not including turbine operation) requiring a scram based on a flux excursion will be terminated sooner with an IRM Range 10 scram than with an APRM scram.

Above the RWM low power setpoint of rated power, the ability of the IRMs to terminate a rod withdrawal transient is limited due to the number and location of IRM detectors. An evaluation was performed that showed by maintaining a minimum core flow of 20.25×10^6 lb/hr (30% rated flow) in range 10, a complete rod withdrawal initiated below 40% of rated power would not result in violating the fuel cladding safety limit. Normal operation of the automatic recirculation pump control will be in excess of 30% rated flow; therefore, little operation below 30% flow is anticipated. Therefore, IRM upscale rod block and scram in range 10 provide adequate protection against a rod withdrawal error transient.

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."

(16) GENE-909-16-0393, "IRM/APRM Overlap Analysis for Nine Mile Point Nuclear Station Unit One," Revision 1, dated April 14, 1993.

(17) GENE-909-39-1093, "IRM/APRM Overlap Improvement for Nine Mile Point Nuclear Station Unit One," dated March 8, 1994.

(18) GENE-C5100196-04, "APRM Flow-Biased Trip Setpoints Stability Long-Term Solution Option II," dated June 1997.

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.

For the APRM scram, the setpoint has been derived based on GE setpoint methodology as outlined in NEDC-31336, "GE Instrumentation Setpoint Methodology." In this methodology, the setpoint is defined as three values, Nominal Trip Setpoint, Allowable Value, and Analytical Limit. The operator will set the Nominal Trip Setpoint. The Allowable Value is listed in the Bases for Specifications 3.6.2 and 4.6.2. The analytical limit is listed in Specification 2.1.2a.

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

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) Verify SRM/IRM channels overlap during startup after the mode switch has been placed in startup. Verify IRM/APRM channels overlap at least 1/2 decade during entry into startup from run (normal shutdown) if not performed within the previous 7 days.

(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 difference is greater than +2.0/-1.9% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 2.1.2a shall not be included in determining the difference.

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

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

Neutron Flux

The APRM scram and rod block setpoints have been derived based on GE setpoint methodology as outlined in NEDC-31336, "GE Instrumentation Setpoint Methodology." In this methodology, the setpoints are defined as three values, Nominal Trip Setpoints, Allowable Values, and Analytical Limits. The analytical limits are listed in Specification 2.1.2a. The allowable values are listed below:

The minimum of:

<u>For W ≥ 0%</u>:

$S \leq (0.55W + 64.46\%)\,T$ with a maximum value of 119.5%
$S_{RB} \leq (0.55W + 59.46\%)\,T$ with a maximum value of 114.5%

AND:

<u>For 14.42% ≤ W ≤ 45%</u>:

$S \leq (1.287W + 16.6\%)$
$S_{RB} \leq (1.287W + 9.312\%)$

WHERE:

S or S_{RB} = The respective scram or rod block allowable value
W = Loop Recirculation Flow as a percentage of the loop recirculation flow which produces a rated core flow of 67.5 MLB/HR
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

IRM, ± 2.5% of rated neutron flux

The APRM downscale rod block setpoint has been derived based on GE setpoint methodology as outlined in NEDC-31336, "GE Instrumentation Setpoint Methodology." In this methodology, the setpoint is defined as three values, Nominal Trip Setpoint, Allowable Value, and Analytical Limit. Table 3.6.2g shows the nominal trip setpoints. The corresponding allowable value is as follows:

APRM Downscale Rod Block, allowable value is ≥[4.24/125] divisions of full scale

Recirculation Flow Upscale, ±1.6% of rated recirculation flow (analytical limit is 107.1% of rated flow)
Recirculation Flow Comparator, ±2.09% of rated recirculation flow (analytical limit is 10% flow differential)

Reactor Pressure, ±15.8 psig

Containment Pressure ±0.053 psig

Reactor Water Level, ±2.6 inches of water

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

Scram Discharge Volume, +0 and -1 gallon

Condenser Low Vacuum, ±0.5 inches of mercury

1) NEDE-24011-P-A "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL" (Latest approved revision).

2) NEDE-30966-P-A "SAFER MODEL FOR EVALUATION OF LOSS-OF-COOLANT ACCIDENTS FOR JET PUMP AND NON-JET PUMP PLANTS" (Latest Approved Revisions)

Vol I "SAFER LONG TERM INVENTORY MODEL FOR BWR LOSS-OF-COOLANT ACCIDENT ANALYSIS"

Vol II "SAFER APPLICATION METHODOLOGY FOR NON-JET PUMP PLANTS"

3) NEDO-20556-P-A "GENERAL ELECTRIC COMPANY ANALYTICAL MODEL FOR LOSS-OF-COOLANT ACCIDENT ANALYSIS IN ACCORDANCE WITH 10CFR50 APPENDIX K". (Latest approved revision)

4) NEDO-32465-A, "REACTOR STABILITY DETECT AND SUPPRESS SOLUTIONS LICENSING BASIS METHODOLOGY FOR RELOAD APPLICATIONS," August 1996.

3. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

4. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.9.2 Fire Protection Program Reports

Noncompliances with the Fire Protection Program (as described in the Final Safety Analysis Report) that adversely affect the ability to achieve and maintain safe shutdown in the event of a fire shall be reported in accordance with the requirements of 10CFR50.72 and 10CFR50.73.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 168 TO FACILITY OPERATING LICENSE NO. DRP-63

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-220

1.0 INTRODUCTION

By letter dated November 16, 1998, as supplemented by letter dated June 21, 1999, Niagara Mohawk Power Corporation (NMPC or the licensee) proposed a license amendment to change the Technical Specifications (TSs) for Nine Mile Point Nuclear Station, Unit 1 (NMP1).

The proposed amendment would revise TSs related to the implementation of systems for the detection and suppression of coupled neutronic/thermal-hydraulic instabilities in the reactor. Average Power Range Monitor (APRM) flow control trip reference cards will initiate a reactor scram to limit the oscillation magnitude at reactor trip so as to limit the associated Critical Power Ratio (CPR) change and, in conjunction with Minimum Critical Power Ratio (MCPR) operating limits, assure compliance with the MCPR safety limit. In addition, the changes would increase the APRM flow biased neutron flux scram and control rod block settings to allow plant operation in the Extended Load Line Limit Analysis (ELLLA) region of the power/flow operating curve. Thus, the proposed changes are in regard to setpoints and calculations for fuel cladding integrity. In TS 2.1.2a, the proposed change would be to the equation for determining the flow biased APRM scram and rod block trip setpoints. In Note (m) of TS Table 3.6.2/4.6.2, the proposed change would be to the calibration range for the APRM channel setpoint. In TS 6.9.1.f, which identifies documents approved by NRC for analytical methods used to determine core operating limits, the proposed change would add "NEDO-32465-A, Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications, August 1996."

In its submittal dated November 16, 1998, NMPC identifies changes to the TS Bases that accompany the above TS changes. In the Bases for TS 2.1.1, the change references new equations in TS 2.1.2a. In the Bases for TS 2.1.2a, the change reflects the new setpoints. In the Bases for TS 2.2.2, the change is to the description of the setpoint methodology that is based upon General Electric (GE) Report NEDC-31336, "GE Instrumentation Setpoint Methodology." In the Bases for TS 3.6.2/4.6.2, the change is to the equations and methodology for determining APRM scram and rod block setpoints.

By letter dated June 21, 1999, NMPC provided additional information in support of the application for amendment. The additional information does not affect the Commission's finding of no significant hazards consideration that was issued in a Federal Register notice (63 FR 71968, December 30, 1998, as corrected by 64 FR 4148, January 27, 1999).

2.0 BACKGROUND

The principal purpose of the proposed amendment is to change the APRM flow biased scram settings to limit the reactor power oscillation magnitude during reactor trip, thereby limiting the associated CPR change and, in conjunction with MCPR operating limits, ensure compliance with the MCPR safety limit.

The change to the APRM flow biased neutron flux scram will be implemented by APRM Flow Control Trip Reference (FCTR) cards and by making the required changes to the APRM flow biased neutron flux scram (and rod block) TS. The FCTR cards implement the Enhanced Option I-A solution, which prevents reactor instabilities by automatically excluding reactor operations in regions of the power/flow operating domain that are susceptible to reactor power instabilities. The changes to the TS implement more restrictive TS flow biased scram trip (and rod block) settings in the low flow regions of the power/flow operating map (i.e., the operating conditions most susceptible to reactor instabilities).

In addition, the amendment would increase the APRM flow biased neutron flux scram and control rod block settings to allow plant operation in the previously approved ELLLA region. NMP1 is currently restricted from full use of the ELLLA region of the power/flow map because of the flow biased rod block setpoints. The NRC staff previously approved operation in the ELLLA region by NMP1 Amendment No. 92, dated March 24, 1987.

Existing NMP1 TS 2.1.2, "Fuel Cladding Integrity," applies to trip settings on automatic protective devices related to variables on which fuel safety limits have been placed. TS 2.1.2a delineates the relationships that establish the flow biased APRM scram and rod block trip settings. The maximum values of the scram and rod block trip settings are currently 120% and 110%, respectively. NMPC has concluded from analyses that none of the postulated accidents would violate the established criteria with a 120% scram trip setting.

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. 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 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 local power range monitor (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.

The current NMP1 APRM system consists of eight identical channels. Each channel is provided with eight inputs from the LPRM system to enable it to compute a core average thermal neutron flux. Additionally, the APRM averaging and trip circuits receive total recirculation flow signals and calculate the Reactor Protection System (scram) and the Reactor Manual Control System (control rod block) trip level setpoints in accordance with a specific relationship between core average power and total recirculation flow.

NMP1 has been identified as a stability long-term solution Option II plant, in which the existing quadrant-based APRM trip system, typical of BWR/2 plants, would initiate a reactor scram for a postulated reactor instability and avoid violating the MCPR safety limit. However, at NMP1, the TS flow biased APRM flux trip setting must be modified for Option II implementation. This change is required to limit the oscillation magnitude at reactor trip, thereby limiting the associated CPR change and, in conjunction with the MCPR operating limits, ensures operation within the MCPR Safety Limit. Specifically, changes to TS 2.1.2, "Fuel Cladding Integrity," are proposed. To implement this TS setting change, the eight APRM analog Flow Bias Trip Units were replaced with GE Nuclear Measurement Analysis and Control (NUMAC) Enhanced Option I-A (E1A) digital FCTR cards. With these FCTR cards in place, a scram setpoint can be established that will meet General Design Criterion 12 of Appendix A to 10 CFR Part 50 for fuel design limit protection.

In the E1A approach, prevention of reactor power instabilities is accomplished by automatically excluding reactor operations in regions of the power/flow operating domain that are susceptible to reactor power instabilities. Region I, the Exclusion Region, defines the limit of power and flow conditions that could result in reactor power instability during steady state operations. With the E1A approach, any event that causes the reactor power-flow trajectory to cross into the exclusion region results in an automatic reactor trip, thereby preventing reactor operation in conditions susceptible to reactor instabilities.

Region II, the Restricted Region, defines the set of power and flow conditions within the reactor licensing basis that could result in reactor power instability if stability controls are not in place. A rod block feature prevents controlled entry into this region unless stability controls are in place. Unintentional entry into Region II results in an alarm, which alerts the operator to exit the region immediately. Within this region, a period-based detection system (PBDS) alerts the operator to an approach to reactor instability. Upon reaching a specified PBDS alarm, the operator is instructed to scram the reactor manually.

Region III, the Monitored Region, defines the operating envelope in which instabilities are only possible under conditions that exceed the reactor licensing basis (e.g., highly skewed power distributions that result in critical power ratio violations). The only requirement for operating in Region III is that a period-based monitoring system be operable. Since this region bounds the area where instabilities are possible, no safety-related stability constraints are required outside Region III.

GENE-A13-00360-02, "Application of Stability Long-Term Solution Option II to Nine Mile Point Nuclear Station Unit 1," demonstrated the application of Option II methodology at NMP1 for operating Cycle 12. GENE-A13-00360-02 was submitted to the NRC by letter dated October 2, 1995. As indicated in GENE-A13-00360-02, "detect and suppress" calculations are performed for two points along the rated rod line consistent with the Boiling Water Reactor Owners Group's (BWROG's) detect and suppress methodology (i.e., NEDO-32465, "BWR Owners' Group Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," dated May 1995). The accepted version, NEDO-32465-A, was issued in August 1996. The calculations start at MCPR Operating Limits along the rated rod line for (1) a five recirculation pump trip to natural circulation (i.e., 24.3% rated core flow) and (2) steady-state operation at 40% core flow. The five recirculation pump trip conservatively represents flow runback transients, including operation with one or two isolated recirculation loops. The 40% core flow case conservatively represents plant startup conditions. In addition, GENE-A13-00360-02 documented the calculation of a revised Restricted Region boundary to be implemented at NMP1.

For Cycle 13, consistent with a 10 CFR Part 21 notification made by GE regarding Safety Limit MCPR evaluations (May 24, 1996), NMPC performed a cycle specific safety limit MCPR calculation that resulted in a safety limit of 1.10 for five-loop and four-loop operation and a safety limit of 1.12 for three-loop operation. NMPC's detect and suppress methodology for NMP1 uses the higher of these safety limits as input to the calculations, resulting in a change to the GENE-A13-00360-02 Cycle 12 limits to Cycle 13 limits. The current MCPR Operating Limit at rated core flow on the rated rod line is less than or equal to 1.26 and the MCPR Operating Limit for steady state operation at 40% core flow on the rated rod line is less than or equal to 2.12. In addition, the Cycle 13 evaluation was based upon a base value for the reload batch size of 200 bundles, rather than the value stated in GENE-A13-00360-02.

The detect and suppress methodology applied to NMP1 was a simplification of the BWROG's detect and suppress methodology (NEDO-32465, May 1995). The NMP1 calculation used a combination of bounding and representative inputs to demonstrate with a deterministic calculation that the final MCPR value at oscillation suppression is greater than the MCPR safety limit. The inputs and assumptions used in the analysis to demonstrate MCPR safety limit protection resulted in restrictions on NMP1 APRM scram trip setpoints and MCPR operating limits for stability Option II implementation. As indicated in GENE-A13-00360-02, the specific restrictions are:

- APRM trip analytical limit at 24.3% flow ≤52.1% power
- APRM trip analytical limit at 40.0% flow ≤72.3% power

The proposed change to the APRM flow biased neutron flux scram TS setpoints, (TS 2.1.2) implements the required settings. By letter dated August 19, 1998, the NRC staff issued a safety evaluation regarding GENE-A13-00360-02 and NMPC's previous submittals responding to Generic Letter 94-02, "Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors." The NRC found that GENE-A13-00360-02 was acceptable for use by NMPC.

The APRM flow-biased trip setpoint is currently being maintained within both the current TSs as well as within the setpoint determined in NRC accepted GENE-A13-00360-02 to preclude instabilities. Accordingly, the FCTR cards are currently operational to implement Stability Solution Option II. The proposed changes will revise the TS APRM flow-biased trip setpoint to be consistent with GENE-A13-00360-02. The MCPR operating limits will be maintained in the Core Operating Limits Report. The margin between the APRM flow biased neutron flux scram and the APRM flow biased control rod block was determined via calculation (GENE-C5100196-04, "APRM Flow-Biased Trip Setpoints Stability Long-Term Solution Option II," dated June 1997).

TS Section 6.9.1.f, “Core Operating Limits Report,” subsection 2, lists the documents that describe the analytical methods used to determine the core operating limits, that have been reviewed and approved by the NRC. NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," August 1996, meets these criteria and, therefore, will be added to Section 6.9.1.f.

3.0 EVALUATION

The proposed amendment would change (1) TS 2.1.2 regarding the flow biased APRM scram and rod block trip settings, (2) Note (m) to Table 4.6.2a concerning channel accuracy, and (3) TS 6.9.1.f. The NRC staff's evaluation of each of these changes is presented in the following subsections.

In its submittal, NMPC also provided associated changes to the TS Bases for sections 2.1.1, 2.1.2, 2.2.2, 3.6.2, and 4.6.2. Pursuant to 10 CFR 50.36(a), TS Bases are not part of the TS (i.e., not an integral part of the operating license). These changes are described below for clarity and completeness.

3.1 TS 2.1.2, “Fuel Cladding Integrity,” (TS page 9)

NMPC proposes to replace the flow biased APRM scram and rod block trip settings in TS 2.1.2.a with the following equations:

“The minimum of:

For $W \geq 0\%$:

$S \leq (0.55W + 67\%)T$ with a maximum value of 122%
$S_{RB} \leq (0.55W + 62\%)T$ with a maximum value of 117%

AND:

For $18\% \leq W \leq 40\%$:

$S \leq (1.287W + 20.83\%)$
$S_{RB} \leq (1.287W + 13.54\%)$”

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

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

NMPC indicated that the proposed changes to the APRM flow biased neutron flux scram setting are needed: 1) to allow plant operation in the ELLLA region, which the NRC staff previously approved by Amendment No. 92 dated March 24, 1987; and 2) to limit the oscillation magnitude at reactor trip in the low flow regions of the power/flow operating map.

The NRC staff has reviewed this request for TS changes and finds that the approved methodologies addressed by GENE-A13-00360-02 and License Amendment No. 92 were applied to the proposed changes to the APRM flow biased neutron flux scram and control rod block settings in the low-flow regions to limit the oscillation magnitude at reactor trip and in the high-flow regions to allow plant operation in the previously approved ELLLA region. The NRC staff also finds that Chapter 15 analyses of the Standard Review Plan (i.e., NUREG-0800) indicate satisfactory results for an increase in the analytical limits of 2 percent for the APRM flow-biased scram and 7 percent for the control rod block. NMPC further justified the selection of the 2 percent increase to the APRM scram analytical limit based upon acceptable results of analysis performed by GE (see Memorandum by D. Hood, "Electronic Mail Regarding Proposed License Amendment on Nine Mile Point Nuclear Station, Unit No. 1," dated August 6, 1999). On the basis of these acceptable results, the NRC staff finds the proposed changes to TS 2.1.2 to be acceptable.

3.2 Bases For TS 2.1.1, "Fuel Cladding - Safety Limit," (TS page 14)

The Bases for TS 2.1.1 currently states, in part, that: "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." In the revised Bases, the word "equation" in this sentence is changed to "equations". This change corresponds to the proposed TS changes stated in 3.1 above.

3.3 Bases For TS 2.1.2, "Fuel Cladding - Limiting Safety System Setting," (TS pages 17, 18, 19, and 22)

The Bases for TS 2.1.2 (see TS page 17) currently states, in part, that: "Analyses [5, 6, 8, 9, 10, 11, 13] demonstrate that with a 120% scram trip setting, none of the abnormal operational transients analyzed violate the fuel safety limit and there is a substantial margin from fuel damage." In the revised Bases, NMPC has added an additional reference, Reference 18, to this sentence. Reference 18 refers to GENE-C5100196-04, "APRM Flow-Biased Trip Setpoints Stability Long-Term Solution Option II," dated June 1997. NMPC also changed the specified scram trip setting of "120%" to "122%."

NMPC has supplemented the Bases for TS 2.1.2 by adding the following new paragraph to TS page 18: "Also, a scram setting has been established to preclude thermal-hydraulic instabilities which could compromise fuel safety limits. Specifically, the scram setting will limit the oscillation magnitude at reactor trip, thereby limiting the associated CPR change, and in conjunction with MCPR operating limits, assure compliance with the MCPR safety limit."

The Bases for TS 2.1.2 currently states, on TS page 18, that "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." NMPC has changed the "110%" in this sentence to "117%."

On TS page 19, the Bases for TS 2.1.2 currently states that "The adequacy of the IRM [Intermediate Range Monitor] scram in range 10 was determined by comparing the scram level on the IRM range 10 to the minimum APRM scram level. The IRM scram is at approximately 38.4% of rated neutron flux while the minimum flow biased APRM scram which occurs at zero recirculation flow is at 65% of rated power." These sentences are changed to state: "The adequacy of the IRM scram in range 10 (approximately 38.4% of rated neutron flux) was determined by comparing the scram level on the IRM range 10 to the minimum APRM scram level for transient protection. The APRM scram level for transient protection is defined by the Section 2.1.2a equation for $W \geq 0\%$. This equation results in a minimum APRM scram of 67% of rated power at zero recirculation flow."

On TS page 22, which is a list of references for Bases 2.1.1 and 2.1.2, the change adds Reference "(18) GENE-C5100196-04, 'APRM Flow-Biased Trip Setpoints Stability Long-Term Solution Option II,' dated June 1997."

3.4 <u>Bases For 2.2.2, "Reactor Coolant System - Limiting Safety System Setting," (TS page26)</u>

Section C of the Bases for 2.2.2 discusses the APRM trip and currently states, in part, that "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." These sentences are changed to state: "For the APRM scram, the setpoint has been derived based on GE setpoint methodology as outlined in NEDC-31336, 'GE Instrumentation Setpoint Methodology.' In this methodology, the setpoint is defined as three values, Nominal Trip Setpoint, Allowable Value, and Analytical Limit. The operator will set the Nominal Trip Setpoint. The Allowable Value is listed in the Bases for Specifications 3.6.2 and 4.6.2. The analytical limit is listed in Specification 2.1.2a."

3.5 <u>Notes For Table 4.6.2a, "Instrumentation That Initiates Scram," (TS page 203)</u>

Table 4.6.2a indicates surveillance requirements for instrumentation that initiates scram. It shows under the neutron flux parameters for "APRM Upscale," an instrument channel calibration is to be performed once per week, and uses Note (m) to amplify this requirement. Note (m) states, in part, "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." The proposed changes would delete the words "absolute" (which occurs twice), and would replace

"2%" with "+2.0/-1.9%." These changes reflect the APRM channel accuracy tolerance which is more conservative. Therefore, the NRC staff finds the proposed changes to be acceptable.

3.6 Bases for 3.6.2 and 4.6.2, "Protective Instrumentation," (TS page 251)

The Bases for 3.6.2 and 4.6.2 currently indicatee the maximum allowable setpoint deviations to be: "APRM Scram, +/- 2.3% of rated neutron flux (analytical limit is 120% of rated flux). APRM Rod Block, +/- 2.3% of rated neutron flux (analytical limit is 110% of rated flux)." This is changed to state: "The APRM scram and rod block setpoints have been derived based on GE setpoint methodology as outlined in NEDC-31336, 'GE Instrumentation Setpoint Methodology.' In this methodology, the setpoints are defined as three values, Nominal Trip Setpoints, Allowable Values, and Analytical Limits. The analytical limits are listed in Specification 2.1.2a. The allowable values are listed below:

The minimum of:

For $W \geq 0\%$:

$S \leq (0.55W + 64.46\%)T$ with a maximum value of 119.5%
$S_{RB} \leq (0.55W + 59.46\%)T$ with a maximum value of 114.5%

AND:

For $14.42\% \leq W \leq 45\%$:

$S \leq (1.287W + 16.6\%)$
$S_{RB} \leq (1.287W + 9.312\%)$

WHERE:

S or S_{RB} = The respective scram or rod block allowable value
W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 67.5 MLB/HR
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 1850MW
CMFLPD = Core Maximum Fraction of Limiting Power Density"

3.7 TS 6.9.1.f, "Reporting Requirements--Routine Reports--Core Operating Limits Report," (TS page 367)

TS Section 6.9.1.f, "Core Operating Limits Report," subsection 2, states that the analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. The subsection currently lists three documents and the proposed change would add a fourth--NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," dated August 1996. The NRC staff finds that this document has been approved previously by the NRC staff and that its addition to the list is appropriate and 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 surveillance requirements. 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 (63 FR 71968, as corrected by 64 FR 4148). 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.

Principal Contributors: M. Waterman
T. Huang
D. Hood

Date: September 21, 1999

DATED: September 21, 1999

AMENDMENT NO. 168 TO FACILITY OPERATING LICENSE NO. DPR-63 NINE MILE POINT NUCLEAR POWER STATION UNIT NO. 1

Docket File
PUBLIC
PDI-1 Reading
S. Bajwa
S. Little
D. Hood
OGC
G. Hill (2), T-5 C3
W. Beckner
ACRS
M. Oprendek, Region I
R. Scholl(e-mail SE only to RFS)
M. Waterman
E. Marinos
T. Huang
J. Wermiel

cc: Plant Service list

Mr. John H. Mueller
Chief Nuclear Officer
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
Operations Building, Second Floor
P.O. Box 63
Lycoming, NY 13093

September 21, 1999

SUBJECT: ISSUANCE OF AMENDMENT FOR NINE MILE POINT NUCLEAR STATION, UNIT NO. 1 (TAC NO. MA4218)

Dear Mr. Mueller:

The Commission has issued the enclosed Amendment No.168 to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station, Unit No. 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated November 16, 1998, as supplemented by letter dated June 21, 1999.

This amendment changes the TSs regarding the setpoints for the average power range monitor flow biased scram to limit the magnitude of reactor power oscillations during a reactor trip, and along with changes to the control rod block settings, allow for operation in the Extended Load Line Limit Analysis region of the power/flow operating curve.

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

Sincerely,

Original signed by:

Darl S. Hood, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-220

Enclosures: 1. Amendment No.168 to DRP-63
2. Safety Evaluation

cc w/encls: See next page

DOCUMENT NAME: G:\PDI-1\NMP1\AMDA4218.WPD

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

OFFICE	PM:PDI	E	LA:PDI		SC:PDI		OGC			
NAME	DHood:lcc		SLittle		SBajwa		R Bachmann			
DATE	9 / 9 /99		9 / 8 /99		9 / 20 /99		9 / 14 /99		/ /99	

Official Record Copy

DATED: September 21, 1999

AMENDMENT NO. 168 TO FACILITY OPERATING LICENSE NO. DPR-63 NINE MILE POINT NUCLEAR POWER STATION UNIT NO. 1

Docket File
PUBLIC
PDI-1 Reading
S. Bajwa
S. Little
D. Hood
OGC
G. Hill (2), T-5 C3
W. Beckner
ACRS
M. Oprendek, Region I
R. Scholl(e-mail SE only to RFS)
M. Waterman
E. Marinos
T. Huang
J. Wermiel

cc: Plant Service list