ATTACHMENT A

NIAGARA MOHAWK POWER CORPORATION
LICENSE NO. DPR-63
DOCKET NO. 50-220

<u>Proposed Changes to Technical Specifications</u>

Replace existing pages 9, 10, 11, 12, 14, 17, 18, 19, 22, 26, 203, 251, and 367 with the attached revised pages. These pages have been retyped in their entirety with marginal markings to indicate changes to the text. Page 251a has been added.

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 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 ≤ 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 ≥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 ≤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.

BASES FOR 2.1.1 FUEL CLADDING - SAFETY LIMIT

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

(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 $\geq$ 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% $\leq$ W $\leq$ 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.

ATTACHMENT B

NIAGARA MOHAWK POWER CORPORATION
LICENSE NO. DPR-63
DOCKET NO. 50-220

Marked-up Copy of Technical Specifications

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 setting shall be established according to the following relationships:

REPLACE WITH INSERT A

$S \leq (0.55W + 65\%)T$ 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

Insert A

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

$S \leq (1.287W + 20.83\%)$

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

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.

S ADD

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] 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.

,18 ADD

122 REVISE

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.

BASES FOR 2.1.2 FUEL CLADDING - LIMITING SAFETY SYSTEM SETTING

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.

REVISE → 117

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% of 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 removed 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.

REPLACE WITH INSERT B

The adequacy of the IRM 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. 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.

Insert B

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.

(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.

ADD (18) GENE-C51001 96-04, "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.

~~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.

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.

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.

DELETE

(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.

+2.0/-1.9 DELETE

REVISE

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 ← REPLACE WITH INSERT C

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)
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

Insert C

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 1850 MW
CMFLPD = Core Maximum Fraction of Limiting Power Density

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-20566-P-A "GENERAL ELECTRIC COMPANY ANALYTICAL MODEL FOR LOSS-OF-COOLANT ACCIDENT ANALYSIS IN ACCORDANCE WITH 10CFR50 APPENDIX K". (Latest approved revision)

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.

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

ADD

ATTACHMENT C

NIAGARA MOHAWK POWER CORPORATION
LICENSE NO. DPR-63
DOCKET NO. 50-220

Supporting Information and No Significant Hazards Consideration Analysis

INTRODUCTION

10CFR50, Appendix A, General Design Criterion (GDC) 10, requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated occurrences. 10CFR50, Appendix A, GDC 12, requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Under certain conditions, Boiling Water Reactors (BWRs) may be susceptible to coupled neutronic/thermal-hydraulic instabilities. These instabilities are characterized by periodic power and flow oscillations and result in density waves (i.e., regions of highly voided coolant periodically sweeping through the core). If the flow and power oscillations become large enough, and the density wave contains a sufficiently high void fraction, then the fuel cladding integrity safety limit could be challenged.

The BWR Owners' Group (BWROG) defined several stability long-term solutions which meet the GDCs stated above. The Option II solution demonstrates that existing quadrant-based Average Power Range Monitor (APRM) trip systems will initiate a reactor scram for a postulated reactor instability and avoid violating the Minimum Critical Power Ratio (MCPR) safety limit. The quadrant-based APRM system is unique to BWR/2 (e.g., NMP1) designs in that Local Power Range Monitor (LPRM) instrument assignments to the APRMs are arranged in separate quadrants of the reactor. Thus, BWR/2s would have a substantial APRM response to a postulated reactor instability which oscillates in either an in-phase (core wide) or out-of-phase (regional) oscillation mode.

Generic Letter 94-02, "Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors," required all BWR licensees to submit a plan to the NRC for their long-term stability corrective actions. A plan acceptable to the NRC would be implementing one of the long-term stability solution options proposed by the BWROG and approved by the NRC. Niagara Mohawk Power Corporation's (NMPC) letter to the NRC dated

December 15, 1995, stated that NMPC would implement the BWROG Option II solution. However, plant-specific analysis for NMP1 (GENE-A13-00360-02, Application of Stability Long-Term Solution Option II to Nine Mile Point Nuclear Station Unit 1) indicated that changes to the APRM flow biased neutron flux scram settings would be required. The changes are required to limit the oscillation magnitude at reactor trip, thereby limiting the associated Critical Power Ratio (CPR) change and, in conjunction with MCPR operating limits, assure compliance with the MCPR safety limit. Accordingly, NMPC has proposed changes to the Technical Specifications (TS) APRM flow biased neutron flux scram setpoint to limit the oscillation magnitude consistent with GENE-A13-00360-02. Also, changes to the APRM flow biased rod block settings are proposed to be consistent with the scram setting changes.

The changes to the APRM flow biased neutron flux scram and rod block TS setpoints will be implemented by APRM Flow Control Trip Reference (FCTR) cards. The FCTR cards replaced the previously installed analog Flow Bias Trip Units under the guidelines of 10CFR50.59. This resulted in the elimination of the first and second levels control rod block alarms. The FCTR cards will implement setpoints in compliance with existing TSs and GENE-A13-00360-02 until this Amendment Application is approved and incorporated.

The proposed change to the scram and control rod block trip reference setpoints increases the complexity of the trip reference function. For this reason, the microprocessor based FCTR card was used to implement and control all card features. To address the stability issue, the proposed changes implement a more restrictive flow biased scram trip setting in the low flow regions of the power/flow operating map (i.e., the operating conditions most susceptible to reactor instabilities). The FCTR cards will provide a scram to ensure that oscillations occurring from steady state operations on the boundary of the exclusion region or after a pump coastdown will not exceed the MCPR safety limit when initiated from the Operating Limit Minimum Critical Power Ratio (OLMPCR). The cards will also establish a margin between the control rod block and neutron flux scram functions. The FCTR cards are similar to the cards described in NEDC-32339P-A, Supplement 2, "Reactor Stability Long-Term Solution: Enhanced Option I-A Solution Design," December 1996. These cards were approved for use for the Enhanced Option I-A solution by the NRC per letter dated September 5, 1995, to the BWROG. NEDC-32696P, Reactor Stability Long-Term Solution: Option II Solution Design, discusses the limited differences between the FCTR cards used in the Enhanced Option I-A solution option and the cards used to implement Option II at NMP1. As indicated in NEDC-32696P, the cards are acceptable for implementing the long term stability solution at NMP1. NEDC-32696P has not been submitted to the NRC.

NMPC also proposes to revise the APRM flow biased neutron flux scram and control rod block TS to provide an increase above their current values in operating conditions not susceptible to reactor instabilities. Specifically, the proposed change will result in a 2% increase and a 7% increase in the analytical limits of the

APRM flow-biased scram and control rod block, respectively. This increase will allow plant operation in the previously approved Extended Load Line Limit Analysis (ELLLA) region.

NMP1 is currently restricted from full use of the ELLLA region of the power/flow map because of the required setpoints for the flow biased rod block. The required setpoint methodology assigns a penalty to account for drift of the power and flow measuring instruments. This reduces the as-left setpoints of the rod block which causes it to fall into the allowable ELLLA region. Consequently, NMP1 is forced to operate at higher core flows to avoid nuisance rod block alarms. Full use of the ELLLA region will allow for optimum core power distributions throughout the operating cycle and a gain in cycle energy. Regaining the full power operating map will save NMP1 approximately $250,000 a year. Operation in the ELLLA region was approved by Amendment No. 92 dated March 24, 1987.

EVALUATION

CURRENT FLOW BIASED APRM SCRAM AND ROD BLOCK TRIP TECHNICAL SPECIFICATION

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. Analyses demonstrate that none of the postulated accidents results in violating 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 flow variable trip setting provides substantial margin from fuel damage, assuming 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.

STABILITY SOLUTION OPTION II, CHANGES TO THE APRM FLOW-BIASED NEUTRON FLUX SCRAM AND CONTROL ROD BLOCK IN THE LOW FLOW REGION OF THE POWER/FLOW MAP

As previously discussed, NMP1 has been identified as a stability long-term solution Option II plant. Option II demonstrated that the existing quadrant-based APRM trip system (that is typical of BWR/2 plants such as NMP1) 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 protection of the MCPR Safety Limit. Specifically, changes to TS 2.1.2, "Fuel Cladding Integrity," will be made. To implement this TS setting change, the eight APRM analog Flow Bias Trip Units were replaced with digital FCTR cards. With the FCTR cards in place, a scram setpoint can be established that will meet GDC 12 criterion for fuel design limit protection.

The current NMP1 APRM System consists of eight (8) identical channels. Each channel is provided with eight (8) inputs from the LPRM System to enable it to compute an accurate core average thermal neutron flux. The averaging and trip circuits of the APRMs receive the total recirculation flow signals and use them to cause the output trip level setpoints to vary in accordance with a specific relationship between core average power and total recirculation flow. The trip signal outputs of the APRMs are utilized in the Reactor Protection System (scram) and the Reactor Manual Control System (control rod block).

GENE-A13-00360-02, Application of Stability Long-Term Solution Option II to Nine Mile Point Nuclear Station Unit 1, provided a demonstration of the application of Option II methodology at NMP1 for 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 BWROG detect and suppress methodology (BWR Owners' Group Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications, NEDO-32465, which was issued in May 1995. The accepted version, NEDO-32465-A was issued in August 1996). The two conditions 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, and 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 Part 21 notification made by GE regarding Safety Limit MCPR evaluations (May 24, 1996), a cycle specific safety limit MCPR calculation was performed resulting in a safety limit of 1.10 for 5 and 4 loop operation and a safety limit of 1.12 for 3 loop operation. NMP1's detect and suppress methodology 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 ≤1.26 and the MCPR Operating Limit for steady state operation at 40% core flow on the rated rod line is ≤2.12. In addition, the Cycle 13 evaluation was based on 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 detect and suppress methodology (NEDO-32465 May 1995). The NMP1 application 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. The NRC's letter dated August 19, 1998 issued the NRC's safety evaluation regarding GENE-A13-00360-02 and our previous submittals responding to Generic Letter 94-02. As indicated in the NRC's letter, GENE-A13-00360-02 was found acceptable for use by NMPC and our responses constitute an acceptable basis for implementing Stability Solution Option II at NMP1. 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 those documents which describe the analytical methods used to determine the core operating limits and which have been previously reviewed and approved by the

NRC. NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," August 1996, meets this criteria and therefore will be added to Section 6.9.1.f.

In summary, the revised TS settings implemented by the FCTR cards will provide more restrictive flow-biased scram and rod block trip settings in the lower flow regions of the power/flow operating map. This will create conservatism within this area of the power/flow operating map and limit the magnitude of oscillations at reactor trip.

CHANGES TO INCREASE APRM FLOW-BIASED NEUTRON FLUX SCRAM AND CONTROL ROD BLOCK SETTINGS

Introduction

As previously discussed, 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. This includes the APRM flow biased scram and rod block trip settings. In addition to the changes to the settings to support NMP1's stability solution, changes are proposed which will increase the APRM flow-biased neutron flux scram and control rod block settings to allow plant operation in the ELLLA region. The proposed changes will result in a 2% increase in the analytical limit of the APRM flow-biased flux scram and a 7% increase in the analytical limit of the APRM flow-biased control rod block. The following evaluation demonstrates that these proposed analytical limit increases have negligible impact on the transient events results for NMP1 as performed in Chapter XV of the NMP1 Updated Final Safety Analysis Report (UFSAR), including the limiting transient events which are reanalyzed each reload. Of the twenty-five (25) transient events analyzed in Section XV of the NMP1 UFSAR, only the Inadvertent Startup of Cold Recirculation Loop event and the Recirculation Flow Controller Malfunction - Increase Flow event have potentially impacted results. The Chapter XV Control Rod Drop Accident, as well as the Turbine Trip with No Bypass at Partial Power, were also evaluated.

The purpose of the high neutron flux scram analytical limit is to limit the neutron flux excursion during postulated anticipated operational occurrences (AOOs) to a level such that the fuel rod mechanical integrity can be maintained. NMP1 UFSAR Section XV was reviewed to assess the potential impact of the proposed 2% increase to this parameter on the responses of the transient events. Twenty-five (25) transient events are analyzed in Section XV (UFSAR Table XV-1) and the analytical value assumed in these analyses for the high neutron flux scram is 120% (UFSAR Table XV-2). Of these events, the following four (4) were identified as limiting AOOs for MCPR consideration:

- Turbine Trip without Bypass
- Feedwater Controller Failure Maximum Demand
- Loss of 100°F Feedwater Heating, and
- Control Rod Withdrawal Error

The first three (3) events are core-wide transients while the last event is a localized bundle power excursion event. These transients, along with the Main Steam Isolation Valve (MSIV) closure with direct (position) scram and the MSIV closure (overpressure) events, are analyzed on a cycle-specific basis as part of the NMP1 reload analysis. The remaining nineteen (19) events are considered not limiting and, therefore, are not analyzed each reload. However, these events were reviewed to determine the effect of the proposed change.

Reload Licensing Analysis Events

Both the Turbine Trip without Bypass and the Feedwater Controller Failure Maximum Demand events are terminated by a reactor scram on turbine stop valve fast closure signal. Therefore, the increase in the high neutron flux scram signal has no impact on the severity of these transient events.

The Loss of 100°F Feedwater Heating event is a subcooling increase event which results in a core thermal power and neutron flux increase. For the last reload analysis for which this event was analyzed (the loss of feedwater heating event was analyzed for cycle 11 reload and not included in subsequent reload analyses because of the non-limiting trend), the peak neutron flux and surface heat flux response were 116% and 115%, respectively. Since the peak neutron flux is below the current scram analytical limit of 120%, no reactor scram was initiated. Therefore, the proposed increase for the APRM flow biased neutron flux scram from 120% to 122% has no adverse impact on the Loss of 100°F Feedwater Heating event.

The Control Rod Withdrawal Error event is typically mitigated by the setpoint of the APRM flow-biased rod block system. The severity of this event is dependent on the rod block analytical limit. However, current analytical practice (i.e., GE design procedures for transient analyses) does not take credit for the function of the flow-biased rod block system when simulating the control rod withdrawal error event for NMP1. Therefore, although this analytical limit is increased by 7% (2% for consistency with the 2% increase in the flow-biased flux scram as well as 5% as the result of the increase in operating domain determined by calculation), this increase has no adverse impact on the Control Rod Withdrawal Error event analysis results. Therefore, from a safety viewpoint, the proposed change in the APRM flow-biased control rod block does not impact transient event responses, as previously analyzed in the NMP1 UFSAR Section XV or in the cycle-specific reload licensing analysis.

The MSIV closure with direct (position) scram event is terminated by a reactor scram signal when the MSIV position switches are at ≤10 percent closed from full open. For this event, the reactor scram signal is normally initiated by the MSIV position switches which occurs prior to the reactor flux exceeding the flux scram setpoint. Therefore, the increase in the high neutron flux scram analytical limit will not affect the response to this event.

NMP1 originally analyzed the MSIV closure (overpressure) event with no scram following closure of all MSIVs. Recent overpressurization analysis was performed for NMP1 simulating reduction in the number of reactor head safety valves from 16 to 9 and taking credit for the reactor flux scram function (TS Amendment No. 152). The proposed 2% increase in the high neutron flux scram analytical limit would result in an increase in the peak vessel pressure to a maximum pressure of 1339 psig (J11-02962SRLR, Supplemental Reload Licensing Report for Nine Mile Point Nuclear Station Unit 1, Reload 14 Cycle 13), which is below the 1375 psig limit for ASME Code requirements for overpressure protection. This increase does not impact NMP1 conformance to the ASME upset criteria for vessel overpressure protection nor the ATWS overpressure criteria.

UFSAR Transient Events

Of the nineteen (19) UFSAR transient events considered not limiting and, therefore, not reanalyzed as part of the NMP1 reload licensing analysis, the following events involve a reactor neutron flux increase. Accordingly, they were reviewed for potential impact due to the proposed high neutron flux scram analytical limit increase:

- Inadvertent Startup of Cold Recirculation Loop
- Recirculation Flow Controller Malfunction - Increase Flow

The combination of increasing recirculation flow and decreasing core inlet enthalpy during an Inadvertent Startup of Cold Recirculation Loop causes reactor power to rise such that the high neutron flux scram signal is reached early in the event. The fuel average surface heat flux also increases from an initial value of 91% of rated to a peak value of about 100% of rated. The proposed 2% increase for the high neutron flux scram would also result in an increase in the fuel average surface heat flux response. However, there is a significant margin between the surface heat flux value response for this event and the current limiting MCPR event (the Feedwater Controller Failure Maximum Demand event). As such, any small change to the fuel surface heat flux response due to the neutron flux scram analytical limit increase would not result in the fuel thermal margin requirements for the Inadvertent Startup of Cold Recirculation Loop to exceed the MCPR limits set by the limiting reload analysis event.

The reactor neutron flux for the Recirculation Flow Controller Malfunction - Increase Flow event also showed an increasing trend from its initial value. However, the peak response for this parameter (about 104% of rated) is significantly below the high neutron flux scram analytical limit. Therefore, this event self-terminates prior to tripping via the current high neutron flux scram analytical limit of 120% or the proposed high neutron flux scram analytical limit of 122%. Accordingly, the proposed increase for the high neutron flux scram analytical limit does not affect the response to this transient event.

UFSAR Accident and Turbine Trip with No Bypass at Partial Power

In addition to the UFSAR AOOs, the potential impact of the proposed increased high neutron flux scram analytical limit on the following events are reviewed:

- Turbine Trip with No Bypass at Partial Power (below steam bypass capacity)
- Control Rod Drop Accident

The Turbine Trip with No Bypass at Partial Power (below the steam bypass capacity) was not included in the NMP1 UFSAR. At this low power level, the automatic scram on turbine stop valve fast closure is bypassed since the pressurization transient is expected to be within the turbine bypass capacity. From a design basis viewpoint, since the turbine bypass operation is not assumed, this event will be terminated by either a high dome pressure scram or a high neutron flux scram signal. Although the high pressure scram will mostly likely be the mitigating scram signal, the proposed increase in the high neutron flux scram analytical limit would have negligible impact on the transient response. At partial power below the steam bypass capacity, the margin between OLMCPR limits required by the K_f multipliers and the transient specific OLMCPR result is sufficient to bound any small increase in the transient responses for events terminated with a high neutron flux scram signal. These partial power transient events are not considered limiting events and, as such, they are bounded by the reload specific limiting MCPR transients.

The Control Rod Drop Accident is included in Chapter XV of the NMP1 UFSAR. As noted in NEDE-24011-P-A, "GESTAR II: General Electric Standard Application for Reactor Fuel," the initial power burst from this event is terminated by the Doppler reactivity feedback while the scram provides the final event termination several seconds later. The 120% APRM scram limit was conservatively chosen. The time delay introduced by the small change in analytical limit will be inconsequential due to the extremely rapid power rise for this event (i.e., the time of scram for a 120% analytical limit vs. a 122% analytical limit is essentially the same).

In summary, the increased APRM flow biased scram and rod block TS settings, implemented by the FCTR cards, will have negligible impact on the transient event or accident previously evaluated for NMP1.

Note (m) to Table 4.6.2a Concerning Channel Accuracy

Existing Note (m) to Table 4.6.2a discusses calibration of the APRM channels. Note (m) requires that the APRM channel be adjusted if the absolute difference is greater than 2% of Rated Thermal Power. As previously discussed, the margin between the APRM flow biased neutron flux scram and APRM flow biased control rod block was determined via calculation. This calculation determined the need to establish a slightly tighter tolerance in regards to adjusting 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. Accordingly, NMPC

proposes to change the 2% value to +2%/-1.9% in Note (m) to Table 4.6.2a. Also, the word "absolute" in Note (m) is used to denote +/- 2%. With the new values (i.e., +2%/-1.9%), the word "absolute" is unnecessary and will be deleted.

Bases for TS 3.6.2/4.6.2 and TS 2.2.2

The Bases for TS 3.6.2 and 4.6.2 (page 251) currently describes the maximum allowable setpoint deviations for neutron flux. NMPC proposes to replace this statement with the formulas used to derive the maximum allowable setpoint deviations for the APRM scram and rod block setpoints; and with words indicating that the APRM scram and rod block setpoints are derived based on the GE setpoint methodology outlined in NEDC-31336, "GE Instrumentation Setpoint Methodology." The proposed wording goes on to state that in this methodology, the setpoints are defined as three values, Nominal Trip Setpoints, Allowable Values, and Analytical Limits. For consistency, similar changes are being made to the Bases for TS 2.2.2 (page 26). In summary, these Bases changes simply provide details of the setpoint methodology currently used as well as specific allowable values.

CONCLUSION

The proposed TS changes, implemented by replacement of the previously installed eight (8) APRM analog Flow Bias Trip Units with digital FCTR cards, will not adversely affect the ability of the RPS and Control Rod Block instrumentation to perform their intended functions. The proposed changes to the APRM scram setpoint will provide MCPR safety limit protection in compliance with GDC 12. The proposed changes to increase the analytical limit of the APRM flow biased flux scram by 2% and the APRM flow biased rod block by 7% has negligible impact on previously evaluated transients or accidents. Consequently, the proposed TS changes will not adversely affect the health and safety of the public and will not be inimical to the common defense and security.

NO SIGNIFICANT HAZARDS CONSIDERATION

10CFR50.91 requires that at the time a licensee requests an amendment, it must provide to the Commission its analysis using the standards in 10CFR50.92 concerning the issue of no significant hazards consideration. Therefore, in accordance with 10CFR50.91, the following analysis has been performed.

The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The APRM neutron monitoring system is not an initiator or a precursor to an accident. The neutron monitoring system monitors the power level of the reactor core and provides automatic core protection signals in the event of a power transient. A Restricted Region will be maintained such that the probability of a

stability event is not increased. Therefore, the proposed TS changes cannot affect the probability of a previously evaluated accident.

The proposed TS changes will revise the APRM flow-biased neutron flux scram TS setting to provide automatic protection to assure that anticipated coupled neutronic/thermal-hydraulic instabilities will not compromise established fuel safety limits. The proposed changes will result in a more restrictive APRM flow-biased scram trip setting in the low flow regions of the power/flow operating map (i.e., operational conditions where reactor instabilities are most probable). In other words, the new settings will provide a scram sooner (at a lower power level) than the existing settings. The associated control rod block setting will also be revised. A margin between the control rod block and flux scram has been determined by calculation.

The proposed changes will also revise the APRM flow-biased neutron flux scram and control rod block TS settings to provide an increase above the current values in operating conditions not susceptible to reactor instabilities. Specifically, the proposed changes will implement a 2% increase in the analytical limit of the APRM flow-biased flux scram and a 7% increase in the analytical limit of the APRM flow-biased control rod block. Evaluation demonstrates that these proposed analytical limit increases have negligible impact on the transient events results for NMP1 as documented in Chapter XV of the NMP1 UFSAR, including the limiting transient events which are reanalyzed each reload. Of the twenty-five (25) transient events analyzed in Section XV of the NMP1 UFSAR, only the Inadvertent Startup of Cold Recirculation Loop event and the Recirculation Flow Controller Malfunction - Increase Flow event have potentially impacted results. The Chapter XV Control Rod Drop Accident as well as the Turbine Trip with No Bypass at Partial Power event were also evaluated.

For the Inadvertent Startup of Cold Recirculation Loop event, the proposed 2% increase in the high neutron flux scram would result in an increase in the fuel average surface heat flux response. However, there is significant margin between the surface heat flux value for this event and the current limiting MCPR event (the Feedwater Controller Failure Maximum Demand event). As such, any small change to the fuel surface heat flux response due to the high neutron flux scram analytical limit increase would not result in the fuel thermal margin requirements for the Inadvertent Startup of Cold Recirculation Loop event to exceed the MCPR limits set by the limiting reload analysis event.

The reactor neutron flux for the Recirculation Flow Controller Malfunction - Increase Flow event also showed an increasing trend from its initial value. However, the peak response for this parameter (104% of rated) is significantly below the high neutron flux scram analytical limit. Accordingly, the proposed increase to the high neutron flux scram analytical limit does not affect the response to this transient event.

The Control Rod Drop Accident is included in Chapter XV of the NMP1 UFSAR. As noted in NEDE-24011-P-A, "GESTAR II: General Electric Standard Application for Reactor Fuel," the initial power burst from this event is terminated by the Doppler reactivity feedback while the scram provides the final event termination several seconds later. The 120% APRM scram limit was conservatively chosen. The time delay introduced by the small change in analytical limit will be inconsequential due to the extremely rapid power rise for this event (i.e., the time of scram for a 120% analytical limit vs. a 122% analytical limit is essentially the same).

The proposed Bases changes to TS 3.6.2/4.6.2 and TS 2.2.2 simply provide details of the setpoint methodology currently used as well as specific allowable values.

Therefore, the proposed TS changes to implement a more restrictive flow-biased scram setting to protect against reactor instabilities and the proposed change to increase the high neutron flux scram and rod block analytical limits do not result in a significant increase in the consequences of an accident previously evaluated.

<u>The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated</u>.

The proposed changes will revise the APRM flow-biased neutron flux scram TS settings to assure anticipated coupled neutronic/thermal-hydraulic instabilities will not compromise established fuel safety limits in the low flow regions of the power/flow operating map as well as revise the associated control rod block settings. These changes also propose a 2% increase in the analytical limit of the APRM flow-biased neutron flux scram and a 7% increase in the analytical limit of the APRM flow-biased control rod block. These changes do not introduce any new accident precursors and do not involve any alterations to plant configurations which could initiate a new or different kind of accident. The proposed changes do not affect the intended function of the APRM system nor do they affect the operation of the system in a way which would create a new or different kind of accident.

Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any previously evaluated.

<u>The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety</u>.

More conservative APRM flow-biased neutron flux scram and control rod block settings will be implemented in the low flow regions of the power/flow operating map. The scram setting change will assure that anticipated coupled neutronic/thermal-hydraulic instabilities will not compromise established fuel safety limits. The proposed changes will also implement a 2% increase in the APRM flow-biased neutron flux scram and a 7% increase in the APRM flow-biased control rod block in those operating regions not susceptible to reactor instabilities. Evaluation

demonstrates that these proposed increases have negligible impact on the transient events or accident results for NMP1. The impacted transient events are either not the limiting MCPR event, the peak response to the event is significantly below the high neutron flux scram analytical limit or in the case of the Control Rod Drop Accident, the time delay introduced by the change will be inconsequential due to the extremely rapid power rise. No other events are adversely affected. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.