ATTACHMENT A

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

Proposed Changes to Technical Specifications (Appendix A)

Replace the existing pages 4c, 11, 20, 63, 64, 64a, 64b, 64c, 65, 66, 67, 68, 69, 69a, 69a1, 69b, 70, 70a, 70b, 70d, and 265. These pages have been retyped in their entirety with marginal marking to indicate the changes to the text. Add new page 265a.

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1.28 Ventilation Exhaust Treatment System

A ventilation exhaust treatment system is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be ventilation exhaust treatment system components.

1.29 Venting

Venting is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during venting. Vent, used in system names, does not imply a venting process.

1.30 Reactor Coolant Leakage

a. Identified Leakage

(1) Leakage into closed systems, such as pump seal or valve packing leaks that are captured, flow metered and conducted to a sump or collecting tank, or

(2) Leakage into the primary containment atmosphere from sources that are both specifically located and known not to be from a through-wall crack in the piping within the reactor coolant pressure boundary.

b. Unidentified Leakage

All other leakage of reactor coolant into the primary containment area.

1.31 Core Operating Limits Report

The CORE OPERATING LIMITS REPORT, its supplements and revisions, is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1f. Plant operation within these operating limits is addressed in individual specifications.

Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of a fuel assembly at the condition of the SLCPR would not produce boiling transition. Thus, although it is not required to establish the safety limit, additional margin exists between the safety limit and the actual occurrence of loss of cladding integrity.

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 operating (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 a maximum LHGR within the limit provided in the Core Operating Limits Report for 8x8, 8x8R, P8x8R and GE8x8EB fuel (Reference 15). At 100% power, this limit is reached with a Maximum Total Peaking Factor (MTPF) of 3.02 for 8x8 fuel, 3.00 for 8x8R and P8x8R fuel, and 2.90 for GE8x8EB fuel. During steady-state operation where the total peaking factor is above 2.90, the equation in Figure 2.1.1 will be used to adjust the flow biased scram and APRM rod block set points.

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

Analyses show that with a bundle flow of $28x10^3$ lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Therefore, due to the 4.56 psi driving head, the bundle flow will be greater than $28x10^3$ lb/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at $28x10^3$ lb/hr

(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) FSAR, Volume II, Appendix E.

(4) FSAR, Second Supplement.

(5) FSAR, Volume II, Appendix E.

(6) FSAR, Second Supplement.

(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) Technical Supplement to Petition to Increase Power Level, dated April 1970.

(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 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."

3.1.7 FUEL RODS

Applicability:

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

Objective:

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

Specification:

a. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value provided in the Core Operating Limits Report. If at any time during power operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded at any node in the core, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR at all nodes in the core is not returned to within the prescribed limits within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until APLHGR at all nodes is within the prescribed limits.

4.1.7 FUEL RODS

Applicability:

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

Objective:

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

Specification:

a. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$25 percent rated thermal power.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
b. <u>Linear Heat Generation Rate (LHGR)</u>	b. <u>Linear Heat Generation Rate (LHGR)</u>
During power operation, the Linear Heat Generation Rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the limiting value provided in the Core Operating Limits Report.	The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$25% rated thermal power.
If at any time during power operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded at any location, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR at all locations is not returned to within the prescribed limits within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until LHGR at all locations is within the prescribed limits.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

c. Minimum Critical Power Ratio (MCPR)

During power operation, the MCPR for all fuel at rated power and flow shall be within the limit provided in the Core Operating Limits Report.

If at any time during power operation it is determined by normal surveillance that the above limit is no longer met, action shall be initiated within 15 minutes to restore operation to within the prescribed limit. If all the operating MCPRs are not returned to within the prescribed limit within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until MCPR is within the prescribed limit.

For core flows other than rated, the MCPR limit shall be the limit identified above times K_f where K_f is provided in the Core Operating Limits Report.

d. Power Flow Relationship During Operation

The power/flow relationship shall not exceed the limiting values shown in Figure 3.1.7.aa.

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

c. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at >25% rated thermal power.

d. Power Flow Relationship

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

e. Partial Loop Operation

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

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

e. Partial Loop Operation

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

When operating with four recirculation loops in operation and the remaining loop unisolated, the reactor may operate at 100 percent of full licensed power level in accordance with Figure 3.1.7aa and an APLHGR not to exceed the applicable limiting values provided in the Core Operating Limits Report for the fuel type.

When operating with four recirculation loops in operation and one loop isolated, the reactor may operate at 100 percent of full licensed power in accordance with Figure 3.1.7aa and an APLHGR not to exceed the applicable limiting values provided in the Core Operating Limits Report for the fuel type, provided the following conditions are met for the isolated loop.

1. Suction valve, discharge valve and discharge bypass valve in the isolated loop shall be in the closed position and the associated motor breakers shall be locked in the open position.

2. Associated pump motor circuit breaker shall be opened and the breaker removed.

If these conditions are not met, core power shall be restricted to 90.5 percent of full licensed power.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

When operating with three recirculation loops in operation and the two remaining loops isolated or unisolated, the reactor may operate at 90% of full licensed power in accordance with Figure 3.1.7aa and an APLHGR not to exceed the applicable limiting values provided in the Core Operating Limits Report for the fuel type.

During 3 loop operation, the limiting MCPR shall be adjusted as described in the Core Operating Limits Report.

Power operation in not permitted with less than three recirculation loops in operation.

If at any time during power operation, it is determined by normal surveillance that the limiting value for APLHGR under one and two isolated loop operation is being exceeded at any node in the core, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR at all nodes in the core is not returned to within the prescribed limits for one and two isolated loop operation within two (2) hours, reactor power reduction shall be initiated at a rate not less than 10 percent per hour until APLHGR at all nodes is within the prescribed limits.

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Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature and the peak local cladding oxidation following the postulated design basis loss-of-coolant accident will not exceed the limits specified in 10CFR50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10CFR50, Appendix K limit. The limiting value for APLHGR is provided in the Core Operating Limits Report. The APLHGR curves in the Core Operating Limits Report are based on calculations using the models described in References 13, 15 and 16.

The Reference 13 and 15 LOCA analyses are sensitive to minimum critical power ratio (MCPR). In the Reference 15, analysis a MCPR value of 1.30 was assumed. If future transient analyses should yield a MCPR limit below this value, the Reference 15 LOCA analysis MCPR value would become limiting. The current MCPR limit is provided in the Core Operating Limits Report. For fuel bundles analyzed with the Reference 13 LOCA methodology, assume MCPR values of 1.30 and 1.36 for five recirculation loop and less than five loop operation respectively.

Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation even if fuel pellet densification is postulated (Reference 12). The LHGR shall be checked daily during reactor operation at ≥ 25% power to determine if fuel burnup or control rod movement has caused changes in power distribution.

Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25%, the reactor will be operating at a minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal-hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial startup testing

of the plant, a MCPR evaluation will be made at the 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluations below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

MCPR limits during operation at other rated conditions are provided in the Core Operating Limits Report. For the case of automatic flow control, the K_f factor is determined such that any automatic increase in power (due to flow control) will always result in arriving at the nominal required MCPR at 100% power. For manual flow control, the K_f is determined such that an inadvertent increase in core flow (i.e., operator error or recirculation pump speed controller failure) would result in arriving at the 99.9% limit MCPR when core flow reaches the maximum possible core flow corresponding to a particular setting of the recirculation pump MG set scoop tube maximum speed control limiting set screws. These screws are to be calibrated and set to a particular value and whenever the plant is operating in manual flow control the K_f defined by that setting of the screws is to be used in the determination of required MCPR. This will assure that the reduction in MCPR associated with an inadvertent flow increase always satisfies the 99.9% requirement. Irrespective of the scoop tube setting, the required MCPR is never allowed to be less than the nominal MCPR (i.e., K_f is never less than unity).

<u>Power/Flow Relationship</u>

The power/flow curve is the locus of critical power as a function of flow from which the occurrence of abnormal operating transients will yield results within defined plant safety limits. Each transient and postulated accident applicable to operation of the plant was analyzed along the power/flow line. The analysis (7, 8, 12, 14) justifies the operating envelope bounded by the power/flow curve as long as other operating limits are satisfied. Operation under the power/flow line is designed to enable the direct ascension to full power within the design basis for the plant.

Partial Loop Operation

The requirements of Specification 3.1.7e for partial loop operation in which the idle loop is isolated, precludes the inadvertent startup of a recirculation pump with a cold leg. However, if these conditions cannot be met, power level is restricted to 90.5 percent power based on current transient analysis (Reference 9). For three loop operation, power level is restricted to 90 percent power based on the Reference 13 and 15 LOCA analyses.

The results of the ECCS calculation are affected by one or more recirculation loops being unisolated and out of service. This is due to the fact that credit is taken for extended nucleate boiling caused by flow coastdown in the unbroken loops. The reduced core flow coastdown following the break results in higher peak clad temperature due to an earlier boiling transition time. The results of the ECCS calculations are also affected by one or more recirculation loops being isolated and out of service. The mass of water in the isolated loops unavailable during blowdown results in an earlier uncovery time for the hot node. This results is an increase in the peak clad temperature.

For fuel bundles analyzed with the methodology used in Reference 13, MAPLHGR shall be reduced as required in the Core Operating Limits Report for 4 and 3 loop operation respectively. For fuel bundles analyzed with the methodology used in References 15 and 16, MAPLHGR shall be reduced as required in the Core Operating Limits Report for both 4 and 3 loop operation.

Partial loop operation and its effect on lower plenum flow distribution is summarized in Reference 11. Since the lower plenum hydraulic design in a non-jet pump reactor is virtually identical to a jet pump reactor, application of these results is justified. Additionally, non-jet pump plants contain a cylindrical baffle plate which surrounds the guide tubes and distributes the impinging water jet and forces flow in a circumferential direction around the outside of the baffle.

Recirculation Loops

Requiring the suction and discharge for at least two (2) recirculation loops to be fully open assures that an adequate flow path exists from the annular region between the pressure vessel wall and the core shroud, to the core region. This provides for communication between those areas, thus assuring that reactor water level instrument readings are indicative of the water level in the core region.

When the reactor vessel is flooded to the level of the main steam line nozzle, communication between the core region and annulus exists above the core to ensure that indicative water level monitoring in the core region exists. When the steam separators and dryer are removed, safety limit 2.1.1d and e requires water level to be higher than 9 feet below minimum normal water level (Elevation 302'9"). This level is above the core shroud elevation which would ensure communication between the core region and annulus thus ensuring indicative water level monitoring in the core region. Therefore, maintaining a recirculation loop in the full open position in these two instances are not necessary to ensure indicative water level monitoring.

References (1) thru (6) intentionally deleted.

(7) "Nine Mile Point Nuclear Power Station Unit 1, Load Line Limit Analysis," NEDO-24012.

(8) Licensing Topical Report GE Boiling Water Reactor Generic Reload Fuel Application, NEDE-24011-P-A, August 1978.

(9) Final Safety Analysis Report, Nine Mile Point Nuclear Station, Niagara Mohawk Power Corporation, June 1967.

(10) NRC Safety Evaluation, Amendment No. 24 to DPR-63 contained in letter from G. Lear, NRC, to D. P. Dise dated May 15, 1978.

(11) "Core Flow Distribution in a GE Boiling Water Reactor as Measured in Quad Cities Unit 1," NEDO-10722A.

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

(13) Loss-of-Coolant Accident Analysis Report for Nine Mile Point Unit 1 Nuclear Power Station, NEDO-24348, Aug. 1981.

(14) GE Boiling Water Reactor Extended Load Line Limit Analysis for Nine Mile Point Unit 1 Cycle 9, NEDC-31126, February 1986.

(15) Nine Mile Point Unit 1, Loss-of-Coolant Accident Analysis, NEDC-31446P, June 1987.

(16) Supplement 1 to Nine Mile Point Generating Station Unit 1 SAFER/CORECOOL/GESTR-LOCA Analysis Report NEDC-31446P-1, Class III, September 1987.

6.9.1 Routine Reports (cont'd)

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

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

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

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

f. CORE OPERATING LIMITS REPORT

Core operating limits (MCPR, LHGR, and APLHGR) shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in NEDE - 24011 "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL", latest approved revision and in the NEDO - 24348 "Loss-of-Coolant Accident Analysis Report for Nine Mile Point Unit 1 Nuclear Power Station" and NEDC 31446 "Nine Mile Point Unit 1 SAFER/CORECOOL/GESTR-LOCA Loss-of-Coolant Accident Analysis" latest revision or supplement. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, 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

in accordance with 10 CFR 50.4 OHB 7/26/88

a. Submit a special report ~~to the appropriate regional office~~ as follows:

Regional Administrator OHB 7/26/88

- Notify the ~~Director~~ of the appropriate Regional Office by telephone within 24 hours.
- Confirm by telegraph, mailgram or facsimile transmission no later than the first working day following the event, and
- Follow-up in writing within 14 days after the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to an operable status.

in accordance with 10CFR50.4 OHB 7/26/88

b. Submit a special report ~~to the Director of the appropriate regional office~~ within 30 days following the event outlining the plans and procedures to be used to restore the inoperable equipment to an operable status.

ATTACHMENT B

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

Supporting Information and No Significant Hazards Consideration Analysis

The proposed changes implement NRC recommendations in Generic Letter 88-16 to remove cycle-specific parameters from the Technical Specifications. Rather this information will be contained in the Core Operating Limits Report which will be submitted to the NRC for information but will not require approval. The cycle-specific parameters to be provided in the Core Operating Limits Report are the Minimum Critical Power Ratio (MCPR), Linear Heat Generation Rate (LHGR) and the Average Planar Linear Heat Generation Rate (APLHGR). Changes in future reload analysis affecting fuel cycle parameters provided in the Core Operating Limits Report will not require license amendments. Revisions of the Core Operating Limits Report will be sent to the NRC for information prior to operation in the cycle for which the report is applicable. The proposed changes follow the NRC guidance contained in Generic Letter 88-16, and consist of three separate actions: (1) the addition of the definition of the Core Operating Limits report that includes the values of cycle-specific parameter limits that have been established using a NRC approved methodology and consistent with all applicable limits of the safety analysis, (2) the addition of an administrative reporting requirement to submit the formal report on cycle-specific parameter limits to the NRC for information and (3) the modification of the individual Technical Specifications to note that cycle-specific parameters shall be maintained within the limits provided in the defined formal report.

The methodology for determining these cycle-specific parameter limits is documented in General Electric's NRC Approved Topical Report, "General Electric Standard Application for Reactor Fuel" (GESTAR) NEDE-24011-P-A, (latest approved revision September 1988) and in the latest revisions or supplements of NEDO-24348 "Loss-of-Coolant Accident Analysis Report for Nine Mile Point Unit 1 Nuclear Power Station" and NEDC-31446 Nine Mile Point Unit 1 SAFER/CORECOOL/GESTR-LOCA Loss-of-Coolant Accident Analysis". Changes to cycle-specific parameters limits are made using this methodology and are consistent with all applicable limits of the safety analysis contained in the Nine Mile Point Unit 1 Updated Final Safety Analysis report (UFSAR). References on page 70d in the bases associated with curves contained in the Core Operating Limits Report have been deleted.

Verification of the use of approved methodology and consistency with the UFSAR will be made in accordance with 10CFR 50.59 for each fuel cycle calculation.

10CFR 50.91 requires that at the time a licensee requests an amendment, it must provide to the Commission its analysis using the standards in Section 50.92 about the issue of no significant hazards consideration. Therefore, in accordance with 10CFR 50.91 and 10 CFR 50.92, the following analysis has been performed:

The operation of Nine Mile Point 1, in accordance with the proposed amendment, will not significantly increase the possibility or consequence of an accident previously evaluated.

This change is administrative in nature as cycle specific parameters will be defined in a controlled document (Core Operating Limits Report) that will be provided to the Commission for information. The Technical Specifications will be amended to require compliance with the applicable limits (MCPR, LHGR, APLHGR) specified in the Core Operating Limits Report. The parameters will be calculated as is presently done using NRC approved methodology. Since the limits are controlled and there is no change in the methods of determining these limits, there is no significant increase in the possibility or consequences of an accident previously evaluated.

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.

The proposed changes are administrative and do not modify plant responses to any operational or accident event. These changes do not create the possibility of a new or different kind of accident.

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.

The proposed changes do not modify or change any operational limit. Core operating limits will continue to be determined by using NRC approved methodology. There will be no impact on the margin of safety as defined in the bases of the Technical Specifications. Consequently, there is no reduction in margins of safety.