ATTACHMENT A

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

Proposed Changed to Technical Specifications (Appendix A)

The existing pages 64b, 64c, 69a, 70, 70b and 70d will be replaced with the attached revised pages. These pages have been reprinted in their entirety with marginal markings to indicate changes to the text.

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

e. Partial Loop Operation

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

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 98 percent of the limiting values shown in Figures 3.1.7a, 3.1.7b, 3.1.7c, 3.1.7d, and 3.1.7e and an APLHGR not to exceed 99% of the limiting values shown in Figure 3.1.7f.

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 98 percent of the limiting values shown in Figures 3.1.7a, 3.1.7b, 3.1.7c, 3.1.7d, 3.1.7e, and an APLHGR not to exceed 99% of the limiting values shown in Figure 3.1.7f, provided the following conditions are met for the isolate 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.

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 96 percent of the limiting values shown in Figures 3.1.7a, 3.1.7b, 3.1.7c, 3.1.7d, and 3.1.7e and an APLHGR not to exceed 99% of the limiting values shown in Figure 3.1.7f.

During 3 loop operation, the limiting MCPR shall be increased by 0.01.

Power operation is 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.

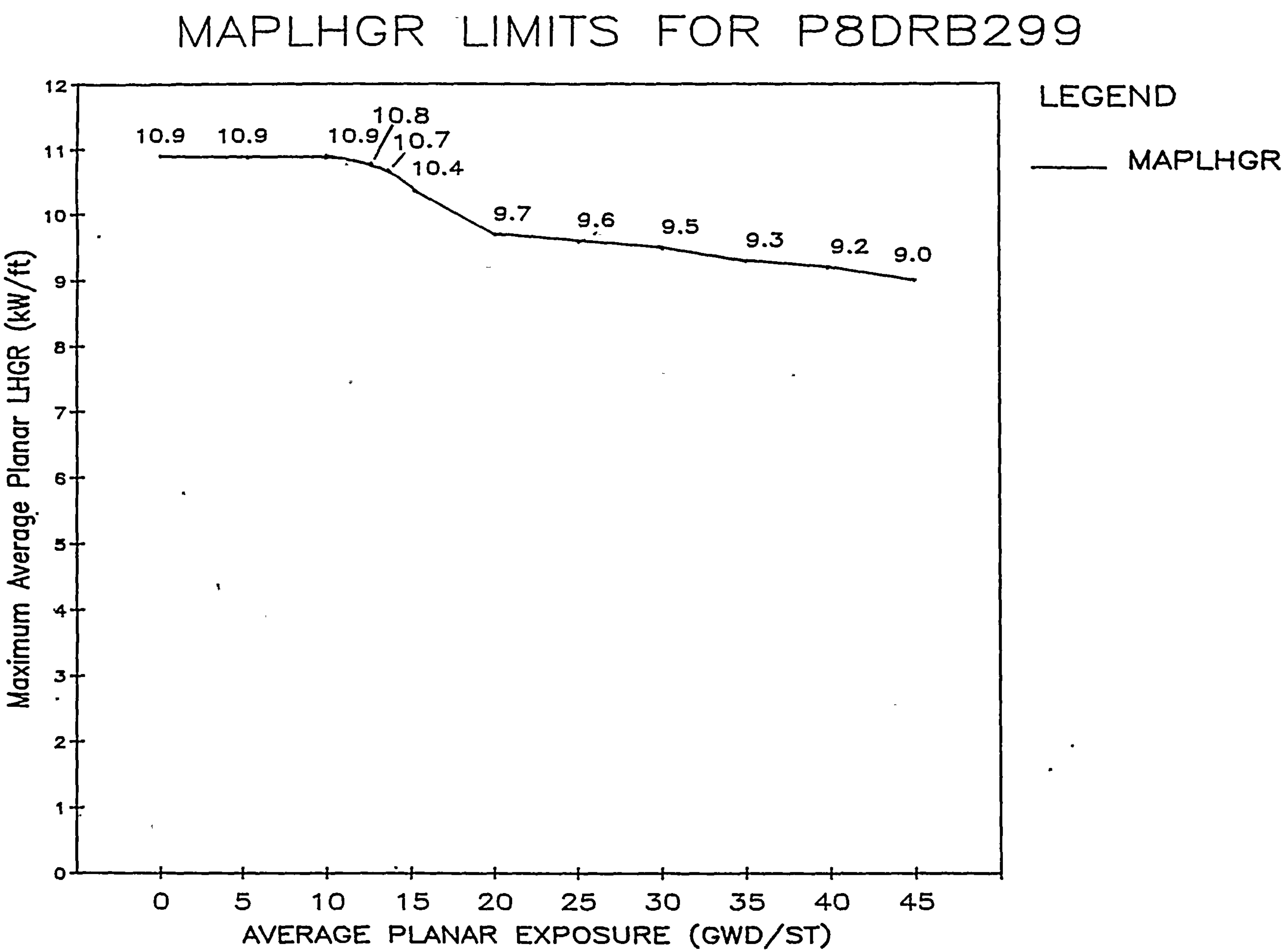

Figure 3.1.7f Maximum Allowable Average Planar LHGR Applicable to P8DRB299 and Future Reload Fuel as described in Reference 15.

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 $\pm$ 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 shown in Figure 3.1.7. These curves are based on calculations using the models described in References 1, 2, 3, 5, 6, 13 and 15.

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 $\geq$ 1.40. For fuel bundles analysed 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 $\geq$ 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

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 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 2% and 4% for 4 and 3 loop operation respectively. For fuel bundles analyzed with the methodology used in Reference 15, MAPLHGR shall be reduced by 1% 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 FOR BASES 3.1.7 AND 4.1.7 FUEL RODS

(1) "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7 and 8, NEDM-10735, August 1973.

(2) Supplement 1 to Technical Report on Densifications of General Electric Reactor Fuels, December 14, 1974 (USAEC Regulatory Staff).

(3) Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

(4) "General Electric Boiling Water Reactor Generic Reload Application for 8 x 8 Fuel," NEDO-20360, Supplement 1 to Revision 1, December 1974.

(5) General Electric Company Analytical Model for Loss of Coolant Analysis in Accordance with 10CFR50 Appendix K," NEDO-20566.

(6) General Electric Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to the USAEC by letter, G.L. Gyorey to Victor Stello, Jr., dated December 20, 1974.

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

(8) Licensing Topical Report General Electric 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 a letter from George Lear, NRC, to D. P. Dise dated May 15, 1978.

(11) "Core Flow Distribution in a General Electric 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 One Nuclear Power Station, NEDO-24348, August 1981.

(14) General Electric 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 One, Loss-of-Coolant Accident Analysis, NEDC-31446P, June 1987.

ATTACHMENT B

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Supporting Information and No Significant Hazards Conditions Analysis

The proposed amendment of Figure 3.1.7(f) and Specification 3.1.7 to the Technical Specifications reflects the use of the SAFER/CORECOOL computer code and methodology to establish the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for the General Electric fuel bundle, type P8DRB299 (Reference 1). Changes to Specification 3.1.7 affect the Limiting Conditions of Operation and Bases for 3.1.7 and 4.1.7 Fuel Rods. Justification for these changes is provided in Report NEDC-31446P (Reference 3, copy attached).

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

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

   The methods used to analyze the Loss of Coolant Accident response of the P8DRB299 fuel conform to 10 CFR 50 Appendix K requirements. The methodology has been approved by the Nuclear Regulatory Commission (Reference 2). The peak cladding temperature was 5°F lower and the maximum oxidation fraction limit was 0.001 lower than that previously calculated for this fuel. Therefore, the proposed amendment will not result in a significant increase in the probability or consequences of an accident previously evaluated.

2. 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 P8DRB299 fuel will still be used. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

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

   An analysis of the Loss of Coolant Accident response of fuel bundle type P8DRB299 has been completed as described in Reference 3 (copy attached) and demonstrates that there is no significant reduction in the margin of safety.

As determined by the analysis above, this proposed amendment has no significant hazards consideration.

REFERENCES

1. Amendment No. 13 to GESTAR-II submitted September 24, 1985.

2. Letter, A. C. Thadani (NRC) to H. C. Pfefferlen (G.E.), "Acceptance for Referencing of Licensing Topical Report NEDE-30996-P, Volume II, 'SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet and Non-Jet Pump Plants,'" May 1987.

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

REFERENCE 3 TO SUPPORTING INFORMATION

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NINE MILE POINT UNIT 1

SAFER/CORECOOL/GESTAR-LOCA
ANALYSIS

NEDC-31446P, JUNE 1987

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