ATTACHMENT A

Niagara Mohawk Power Corporation

License No. DPR-63

Docket No. 50-220

Proposed Changes to Technical Specifications (Appendix A)

Replace pages 64b, 64c, 65, 66, 67, 68, 70, 70b, and 70d with the attached revised pages. Pages 64b, 64c, 70, 70b and 70d have been retyped in their entirety and the marginal markings indicate changes to the text.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

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.

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, and 3.1.7e, 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.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
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 percent 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. 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 determinned by normal surveillance that the limiting value for APLHGR under one and two isolated loop operation is being exceed 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.	

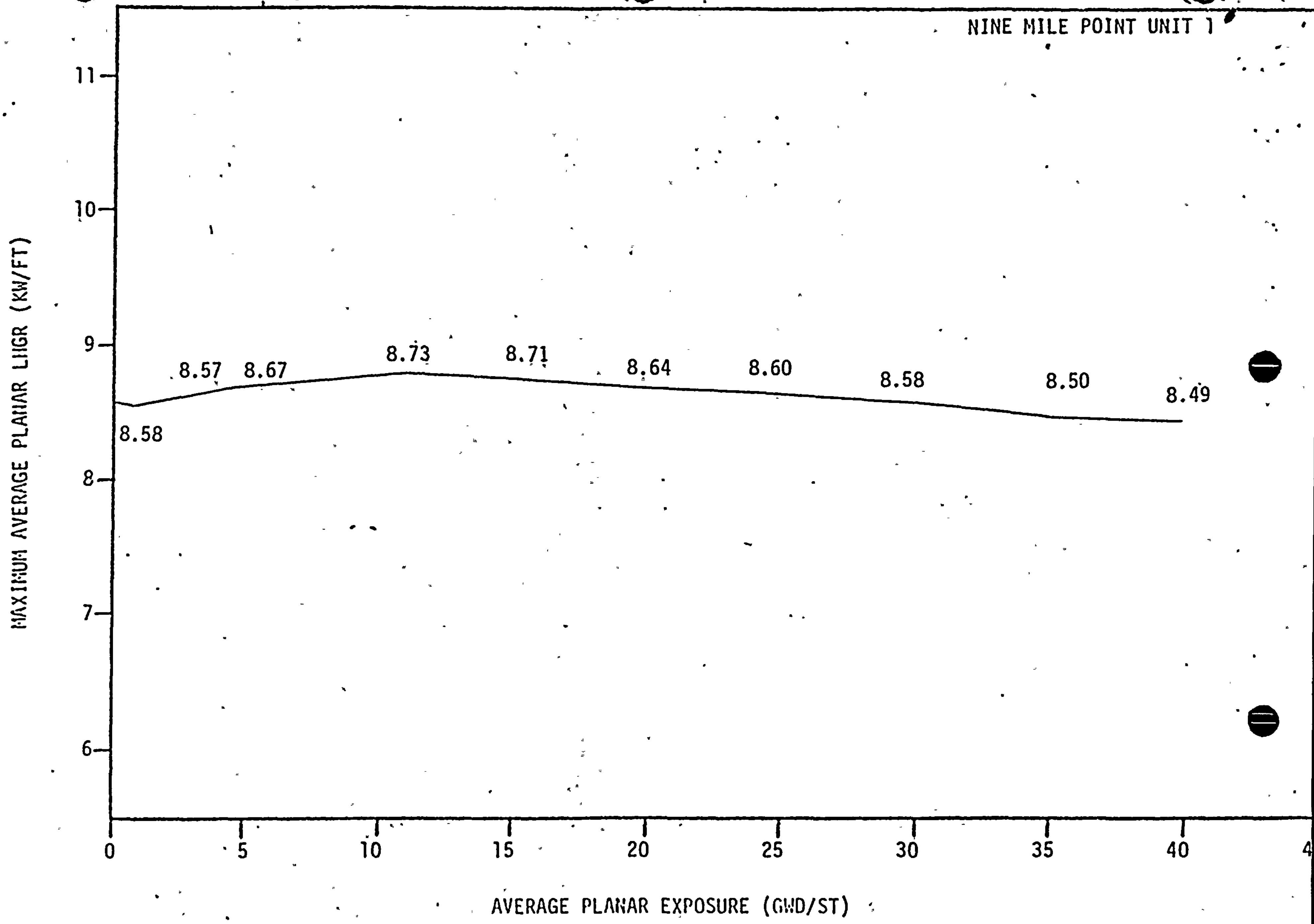

Figure 3.1.7a MAXIMUM ALLOWABLE AVERAGE PLANAR LHGR APPLICABLE TO 8DB250 FUEL AS DESCRIBED IN REFERENCE 8.

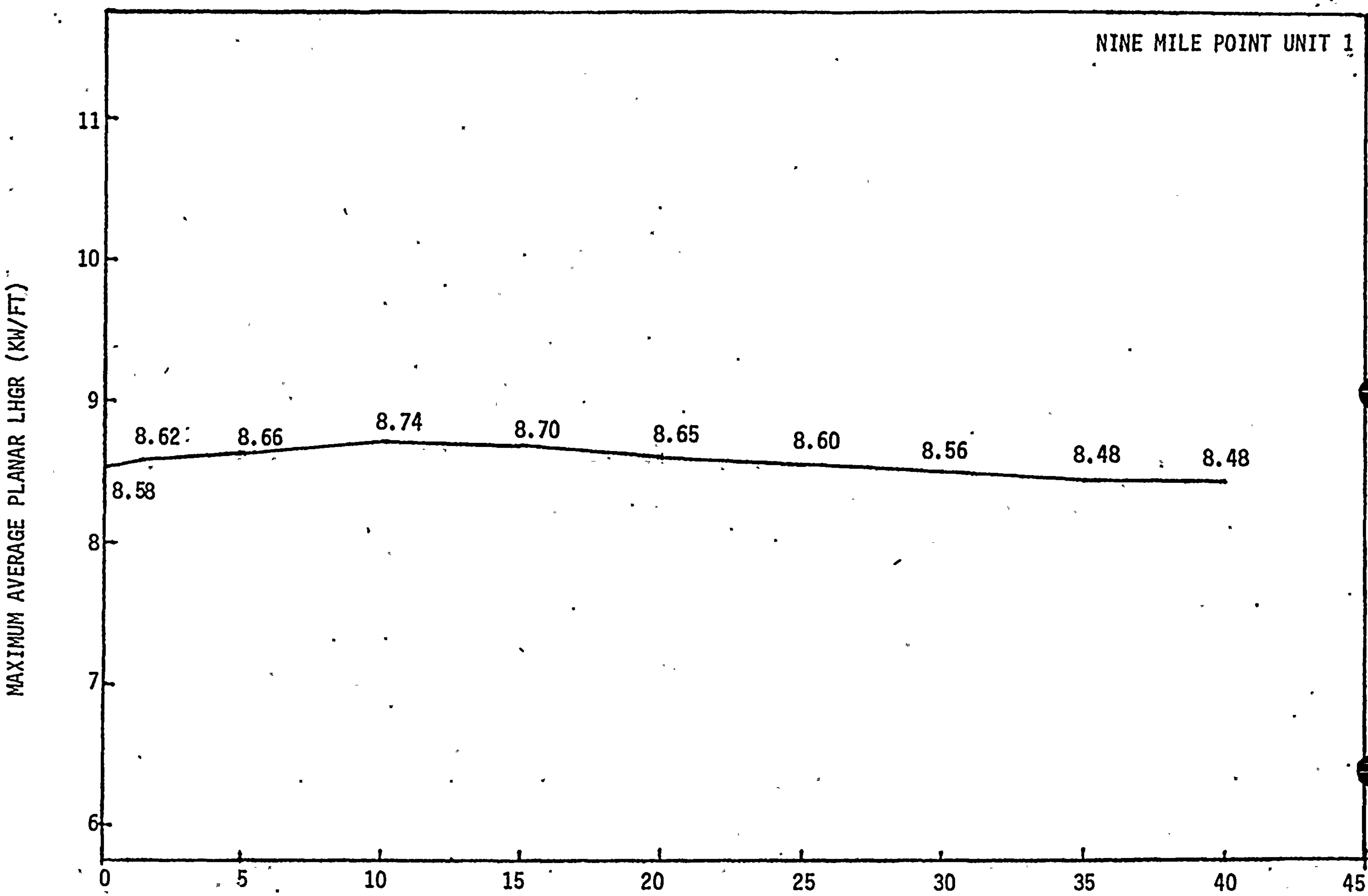

Figure 3.1.7b Maximum Allowable Average Planar LHGR Applicable to 8DB274L and 8DB274H Fuel as described in Reference 8.

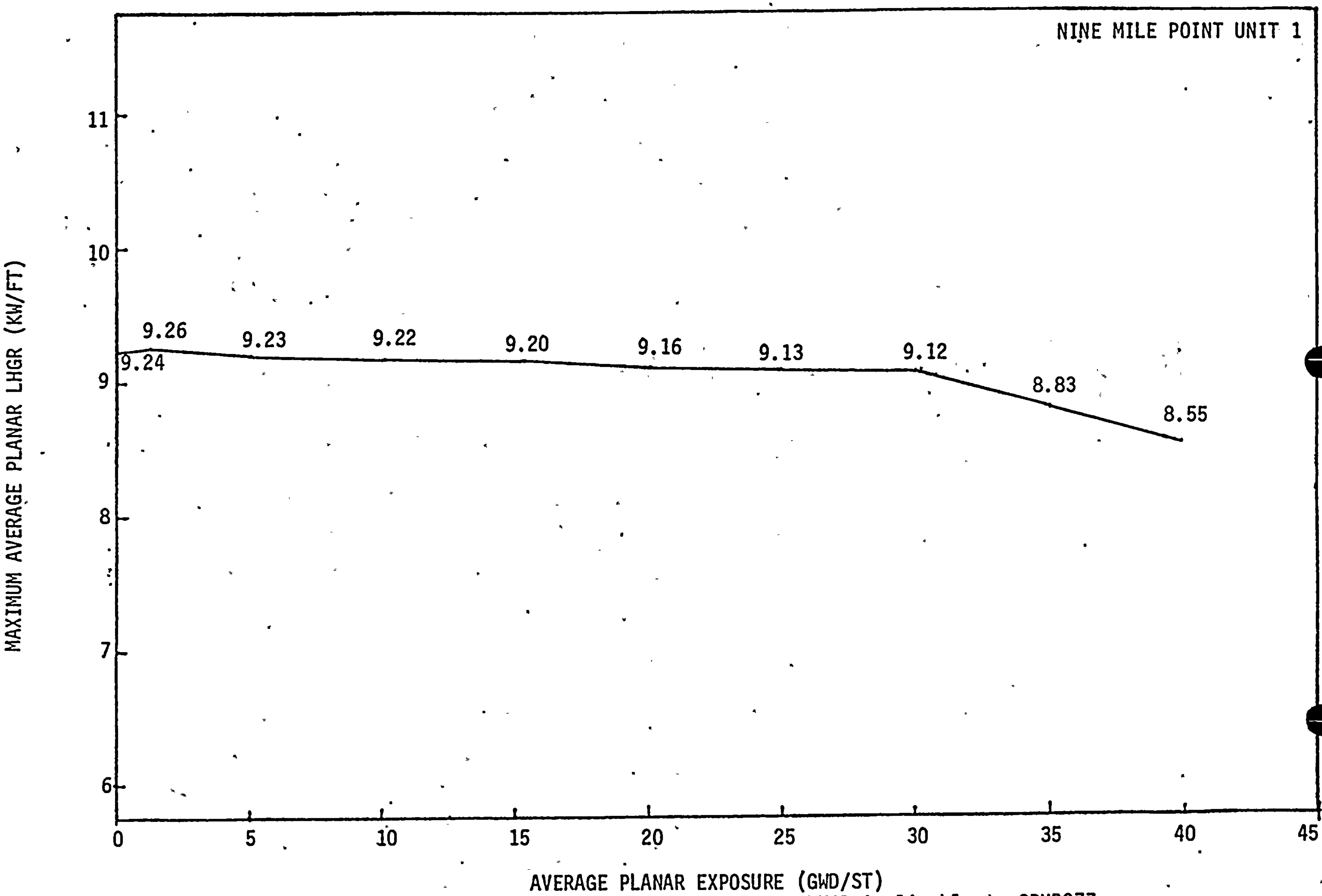

Figure 3.1.7c Maximum Allowable Average Planar LHGR Applicable to 8DNB277 Fuel as described in Reference 8.

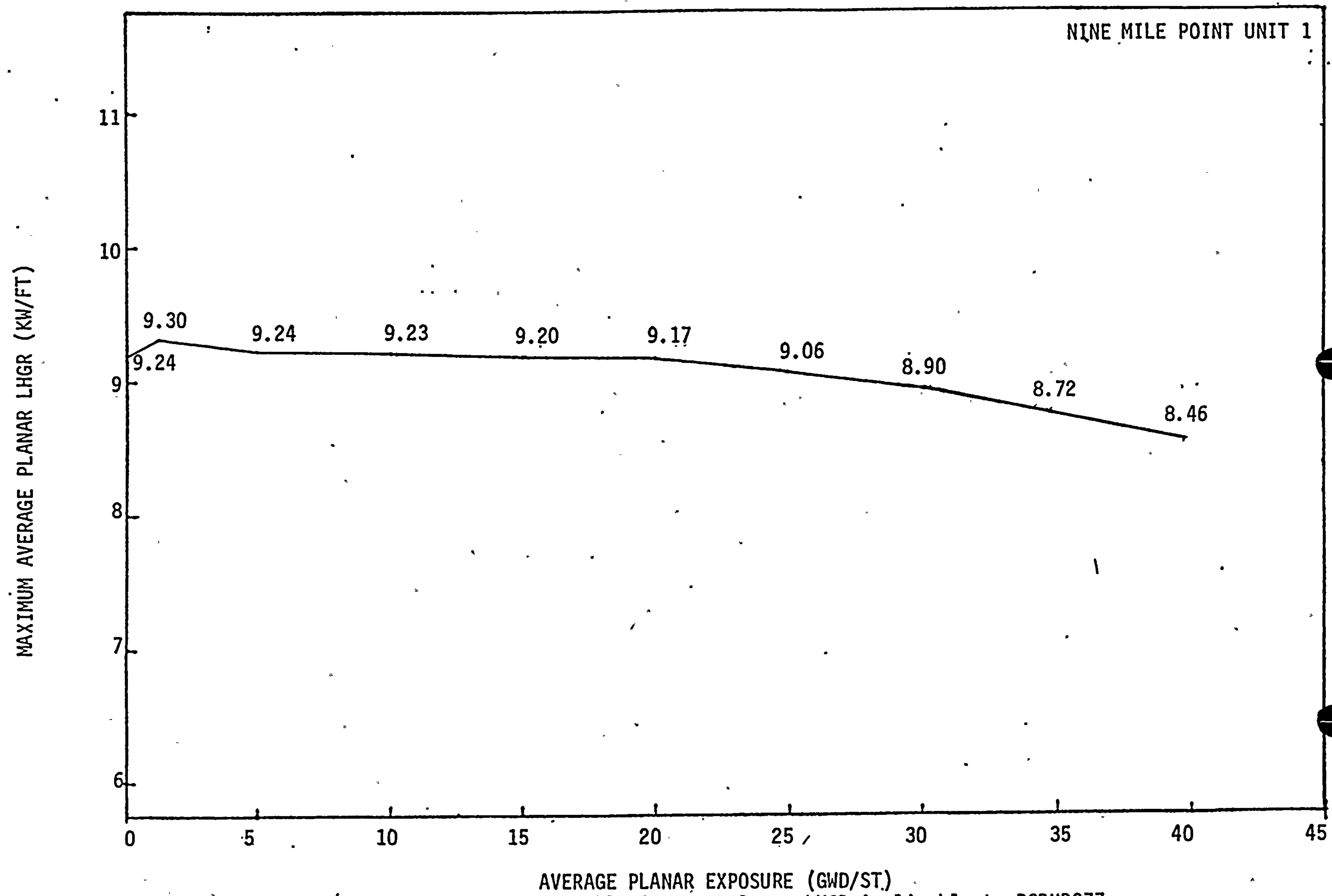

Figure 3.1.7d Maximum Allowable Average Planar LHGR Applicable to P8DNB277 and Future Reload Fuel as described in Reference 8.

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

The Reference 13 LOCA analysis is sensitive to minimum critical power ratio (MCPR). In that analysis MCPR values of 1.30 for 5 loop operation and 1.36 for 4 and 3 loop operation, were assumed. If future transient analyses should yield a MCPR limit below either of these values the Reference 13 LOCA analysis MCPR value would become limiting. The current MCPR limit is 1.38.

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

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 LOCA analysis.

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 a earlier uncovery time for the hot node. This results in an increase in the peak clad temperature.

To assure peak clad temperatures remain below 2200°F, analysis has shown that the limiting average planar linear heat generation rate for each fuel type shall be reduced 2 percent and 4 percent for 4 and 3 loop operation respectively (Reference 13).

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 full 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.

ATTACHMENT B

Niagara Mohawk Power Corporation

License No. DPR-63

Docket No. 50-220

Supporting Information

Attachment A describes proposed changes to the Nine Mile Point Unit 1 Technical Specifications. These changes are required to provide relief from Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) related power derates and more flexibility in operation. The bases for the proposed Technical Specification changes are provided in the enclosed report "Loss-of-Coolant Accident Analysis Report for Nine Mile Point Unit 1 Nuclear Power Station," NEDO-24348.

The present MAPLHGR limits were established by previous analysis (Reference 1). The NRC Safety Evaluation of that analysis (Reference 2) concluded that the MAPLHGR limits were unrealistic but conservative due to the lack of consideration of the extended nucleate boiling anticipated for smaller breaks. The Nine Mile Point Unit 1 loss of coolant accident performance for the entire break spectrum has been reanalyzed to improve MAPLHGR limits by removing the overly conservative heat transfer assumptions employed in the previous analysis.

MAPLHGR limit curves for 8DB250, 8DB274L and 8DB274H fuel types are revised to remove fission gas MAPLHGR reduction factors at high exposures. These penalties are deleted based on References 3 and 4. In these General Electric has demonstrated sufficient peak clad temperature (PCT) margin due to approved ECCS model improvements not taken credit for. These improvements compensate for PCT increase due to enhanced fission gas release at high exposures. That evaluation addressed the NRC fission gas release concerns and provided justification for elimination of any penalties for MAPLHGR limits calculated with current models.

The LOCA performance of fuel types 8DNB277 and P8DNB277 was reanalyzed per the enclosed report. As a result, these fuel types required reevaluation of the PCT margins available to compensate for enhanced fission gas release effects. This reevaluation indicated the need to apply fission gas MAPLHGR reduction factors to the limits developed in the attached reanalysis. Reduction factors were found to be needed at 35 and 40 GWD/ST exposures. The same method as was used in References 3 and 4 was used to develop these reduction factors which are included in the revised MAPLHGR limit curves for these two fuel types.

REFERENCES

1) Nine Mile Point Unit 1 Nuclear Power Station Loss of Coolant Accident Analyses Conformance with 10 CFR 50 Appendix K (Non Jet Pump Plant), October, 1975.

2) Letter, K. R. Goller (NRC) to G. K. Rhode (Niagara Mohawk Power Corporation) "Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 5 to Facility Operating License No. DPR-63," Niagara Mohawk Power Corporation Docket No. 50-220, November 13, 1975.

3) Ronald E. Engel (GE) to T. A. Ippolito (NRC), "Extension of Emergency Core Cooling System Performance Limits," May 6, 1981.

4) Ronald E. Engel (GE) to T. A. Ippolito (NRC), "Additional Information Regarding Extension of Emergency Core Cooling System Performance Limits," May 28, 1981.