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

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 41
License No. DPR-63

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Niagara Mohawk Power Corporation (the licensee) dated April 21, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-63 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 41 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

T.A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 19, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 41

FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

Revise Appendix A as follows:

Remove	Insert
5	5
8	8
10	10
11	11
12	12
16	16
17	17
63	63
64	64
64a	64a
64c	64c
64e	64e
65	65
66	66
67	67
68	68
69	69
70	70
70a	70a
70d	70d
237a	237a
245	245
-	245-1
248	248
-	248-1

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 claddi[illegible] 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 trip settings shall be less than or equal to that shown in Figure 2.1.1.

b. The IRM scram trip setting shall not exceed 12% of rated neutron flux.

c. The reactor high pressure scram trip setting shall be $\leq$ 1080 psig.

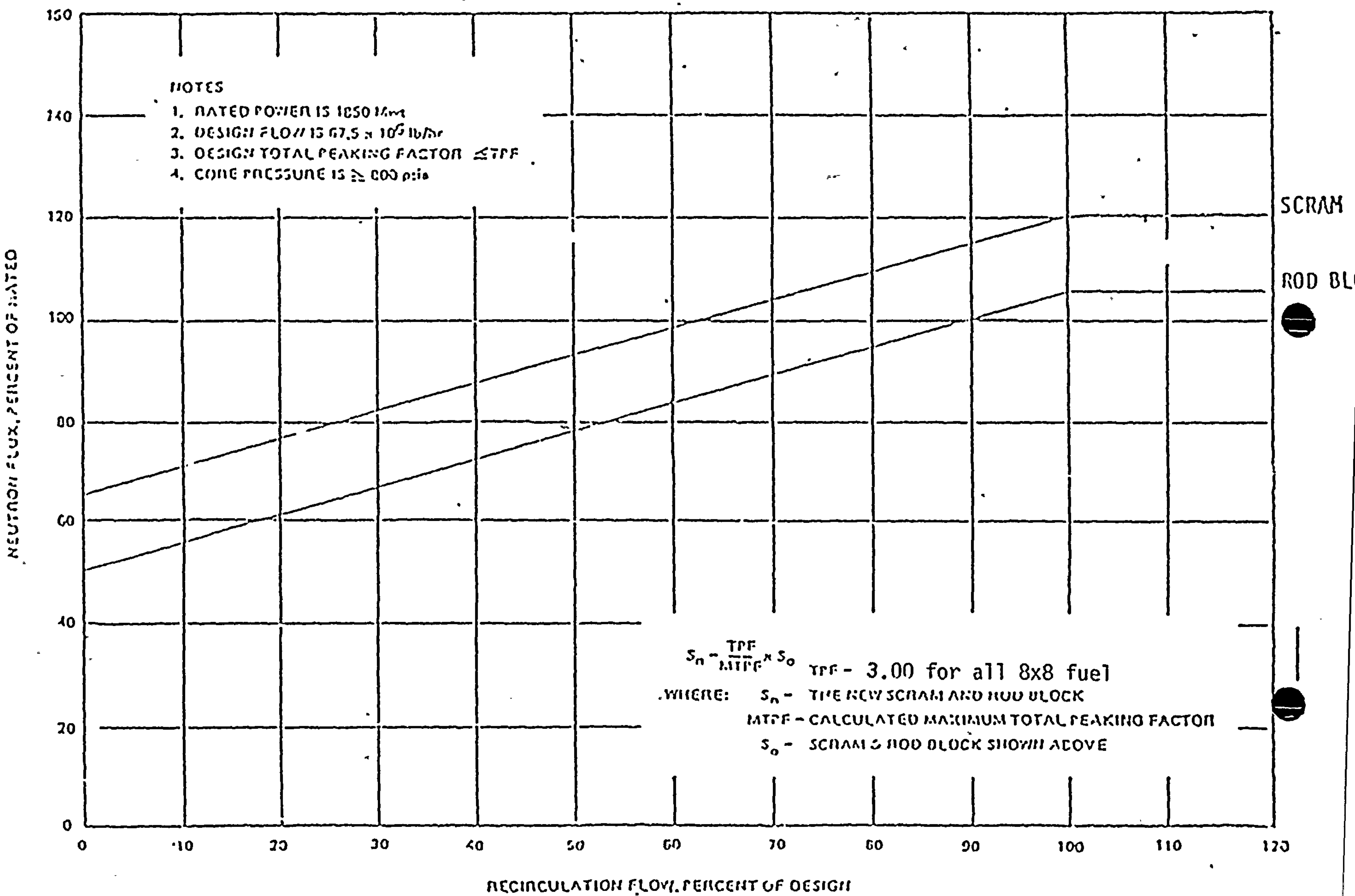

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Figure 2.1.1. Flow Biased Scram and APRM Rod Block

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a safety limit such that the Minimum Critical Power Ratio (MCPR) is no less than the Safety Limit Critical Power Ratio (SLCPR) (Reference 12). The SLCPR represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use-related cracking, the thermally caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding safety limit is defined with margin to the conditions which would produce onset of transition boiling, (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation.

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, at reactor pressure > 800 psia and core flow > 10% of rated the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the Critical Power Ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the Minimum Critical Power Ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective set points via the instrumented variables, by the nominal expected flow control line. The SLCPR has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from a normal operating condition more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the SLCPR is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in References 1 and 12.

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 of 13.4 kW/ft for 8x8 fuel and 13.4 kW/ft for 8x8R fuel. At 100% power this limit is reached with a Maximum Total Peaking Factor (MTPF) of 3.02 for 8x8 fuel and 3.00 for 8x8R fuel. For the case of the MTPF exceeding these values, operation is permitted only at less than 100% of rated thermal power and only with reduced APRM scram settings as required by Specification 2.1.2.a. (In cases where for a short period the total peaking factor was above 3.02 for 8x8 fuel and 3.00 for 8x8R fuel 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

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.

During transient operation the heat flux (thermal power-to-water) would lag behind the neutron flux due to the inherent heat transfer time constant of the fuel which is 8 to 9 seconds. Also, the limiting safety system scram settings are at values which will not allow the reactor to be operated above the safety limit during normal operation or during other plant operating situations which have been analyzed in detail.(3,4) In addition, control rod scrams are such that for normal operating transients the neutron flux transient is terminated before a significant increase in surface heat flux occurs. Scram times of each control rod are checked periodically to assume adequate insertion times. Exceeding a neutron flux scram setting and a failure of the control rods to reduce flux to less than the scram setting within 1.5 seconds does not necessarily imply that fuel is damaged; however, for this specification a safety limit violation will be assumed any time a neutron flux scram setting is exceeded for longer than 1.5 seconds.

If the scram occurs such that the neutron flux dwell time above the limiting safety system setting is less than 1.7 seconds, the safety limit will not be exceeded for normal turbine or generator trips, which are the most severe normal operating transients expected. These analyses show that even if the bypass system fails to operate, the design limit of the SLCPR is not exceeded. Thus, use of a 1.5-second limit provides additional margin.

The process computer has a sequence annunciation program which will indicate the sequence in which scrams occur such as neutron flux, pressure, etc. This program also indicates when the scram set point is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 2.1.1.c will be relied on to determine if a safety limit has been violated.

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 up to 20% flow. This is accomplished by keeping the reactor mode switch in the startup position until 20% flow is exceeded and the APRM's are on scale. Then the reactor mode switch may be switched to the run mode, thereby switching scram protection from the IRM to the APRM system.

In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to 1% of rated power, thus maintaining a limit above the SLCPR. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

c. As demonstrated in Appendix E-I* and the Technical Supplement to Petition to Increase Power Level, the reactor high pressure scram is a backup to the neutron flux scram, turbine stop valve closure scram, generator load rejection scram, and main steam isolation valve closure

scram, for various reactor isolation incidents. However, rapid isolation at lower power levels generally results in high pressure scram preceding other scrams because the transients are slower and those trips associated with the turbine generator are bypassed.

The operator will set the trip setting at 1080 psig or lower. However, the actual set point can be as much as 15.8 psi above the 1080 psig indicated set point due to the deviations discussed above.

d. A reactor water low level scram trip setting -12 inches (53 inches indicator scale) relative to the minimum normal water level (Elevation 302' 9") will assure that power production will be terminated with adequate coolant remaining in the core. The analysis of the feedwater pump loss in the Technical Supplement to Petition to Increase Power Level, dated April 1970, has demonstrated that approximately 4 feet of water remains above the core following the low level scram.

The operator will set the low level trip setting no lower than -12 inches relative to the lowest normal operating level. However, the actual set point can be as much as 2.6 inches lower due to the deviations discussed above.

e. A reactor water low-low level signal -5 feet (5 inches indicator scale) relative to the minimxm normal water level (Elevation 302' 9") will assure that core cooling will continue even if level is dropping. Core spray cooling will adequately cool the core, as discussed in LCO 3.1.4.

The operator will set the low-low level core spray initiation point at no less than -5 feet (5 inches indicator scale) relative to the minimum normal water level (Elevation 302' 9"). However, the actual set point can be as much as 2.6 inches lower due to the deviations discussed above.

f. 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 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

LIMITING CONDITION FOR OPERATION

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 shown in Figures 3.1.7a, 3.1.7b, 3.1.7c, 3.1.7d and 3.1.7e. 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.

SURVEILLANCE REQUIREMENT

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% rated thermal power.

LIMITING CONDITION FOR OPERATION

b. Linear Heat Generation Rate (LHGR)

During power operation, the Linear Heat Generation Rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed 13.4 KW/FT.

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.

SURVEILLANCE REQUIREMENT

b. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked daily during reactor operation at ≥25% rated thermal power.

c. Minimum Critical Power Ratio (MCPR)

During power operation, the MCPR for all 8 x 8 fuel at rated power and flow shall be as shown in the table below:

LIMITING CONDITION FOR OPERATING MCPR

Core Average Incremental Exposure	Limiting MCPR*
BOC to EOC minus 2 GWD/ST	≥ 1.38
EOC minus 2 GWD/ST to EOC minus 1 GWD/ST	≥ 1.41
EOC minus 1 GWD/ST to EOC	≥ 1.50

If at any time during power operation it is determined by normal surveillance that these limits are no longer met, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If all the operating MCPRs are 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 MCPR is within the prescribed limits.

For core flows other than rated the MCPR limits shall be the limits identified above times K_f where K_f is as shown in Figure 3.1.7-1.

d. Power Flow Relationship During Power Operation

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

*These limits shall be determined to be applicable each operating cycle by analyses performed utilizing the ODYN transient code.

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
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, the reactor may operate at 100 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 and 3.1.7c, provided conditions 1 and 2 above are met for the isolated loops. If these conditions are not met, core power shall be restricted to 90.5 percent of full licensed power. 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.	

Fufc.

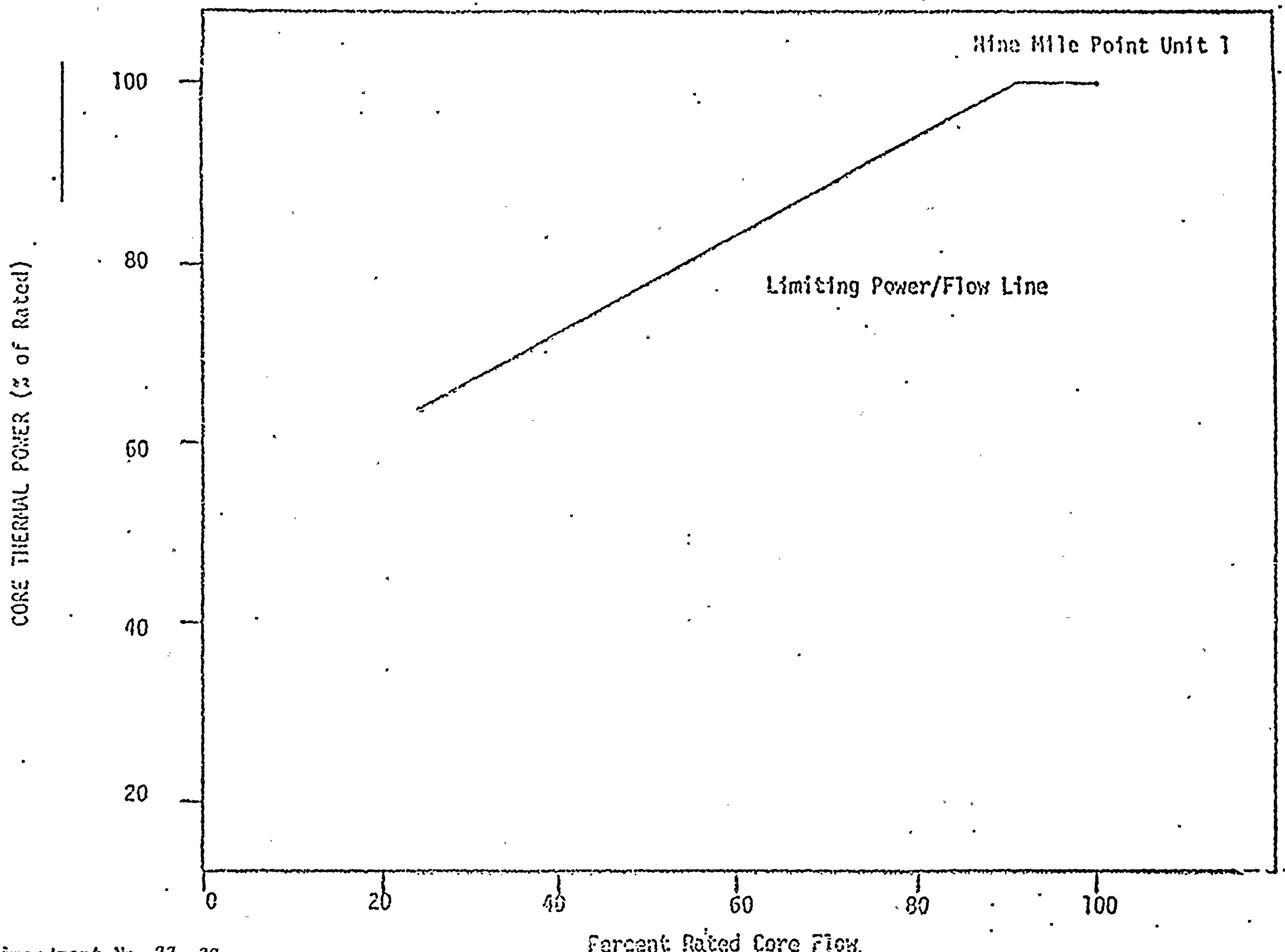

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Figure 3.1.7.aa LIMITING POWER FLOW LINE

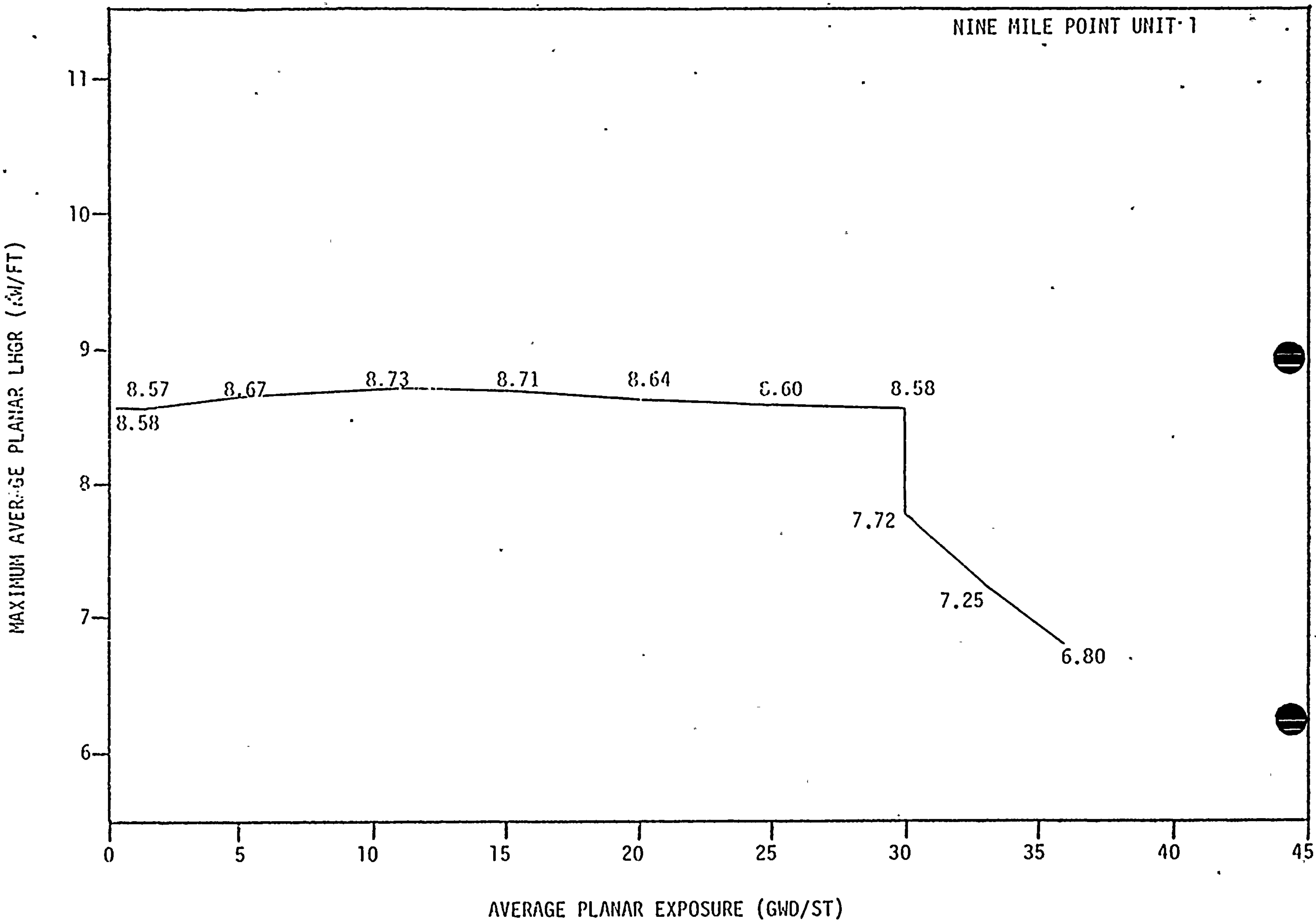

Figure 3.1.7a Maximum Allowable Average Planar LHGR Applicable to 8DB250 Fuel as described in Reference 8

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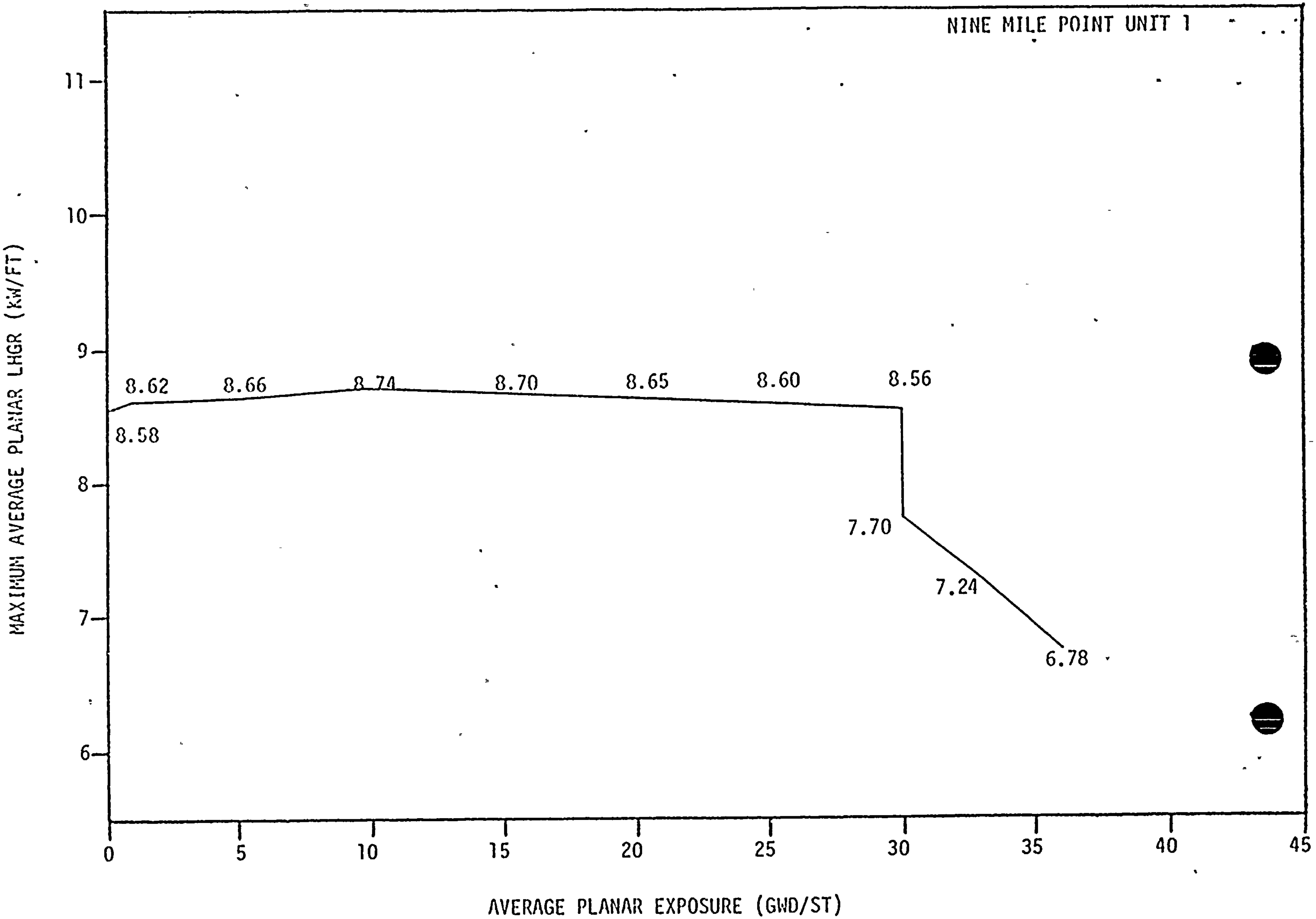

Figure 3.1.7b Maximum Allowable Average Planar LHGR Applicable to 8DB274L and 8DB274H Fuel as described in Reference 8.

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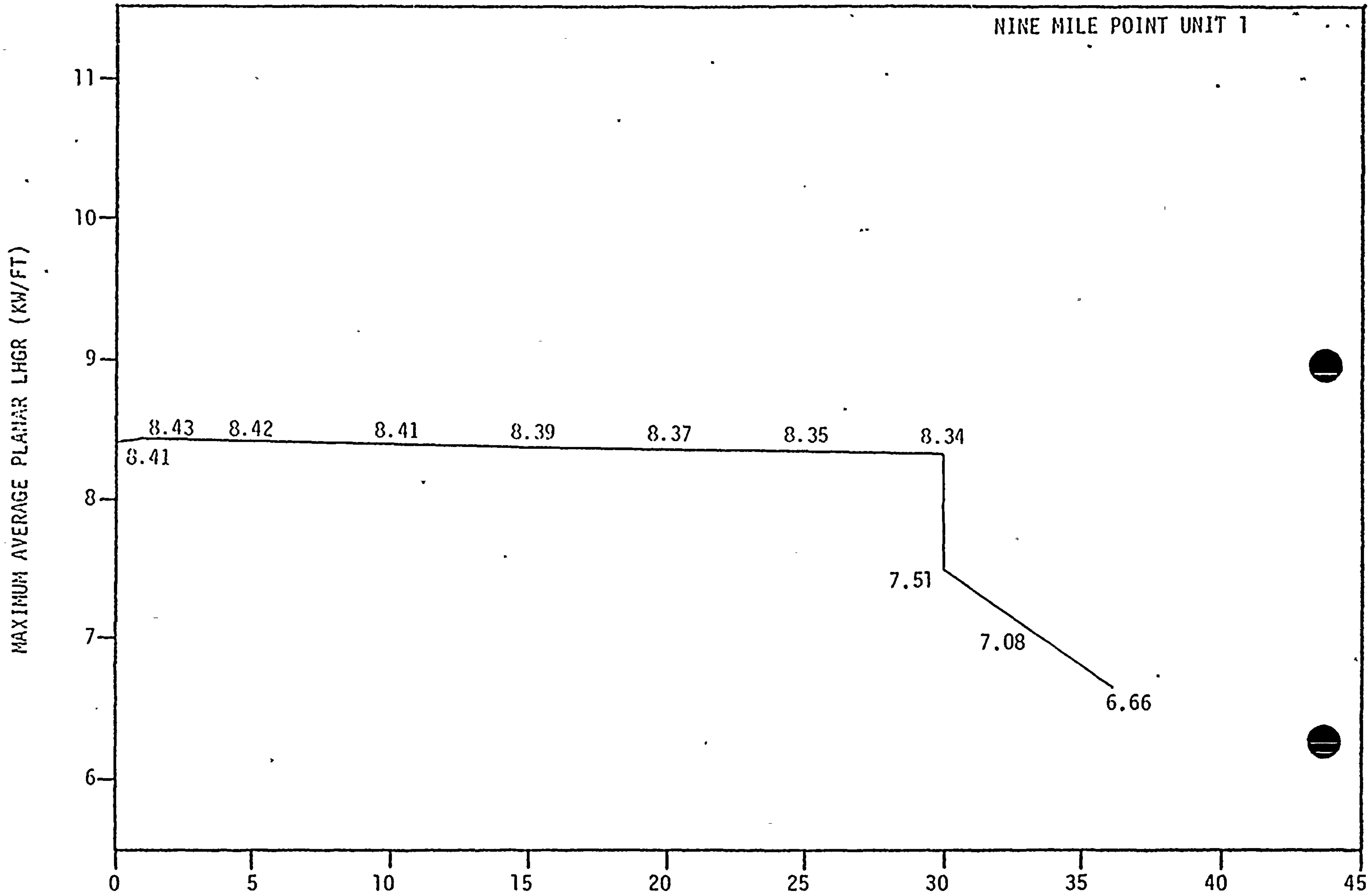

Figure 3.1.7c Maximum Allowable Average Planar LHGR Applicable to 8DNB277 Fuel as described in Reference 8

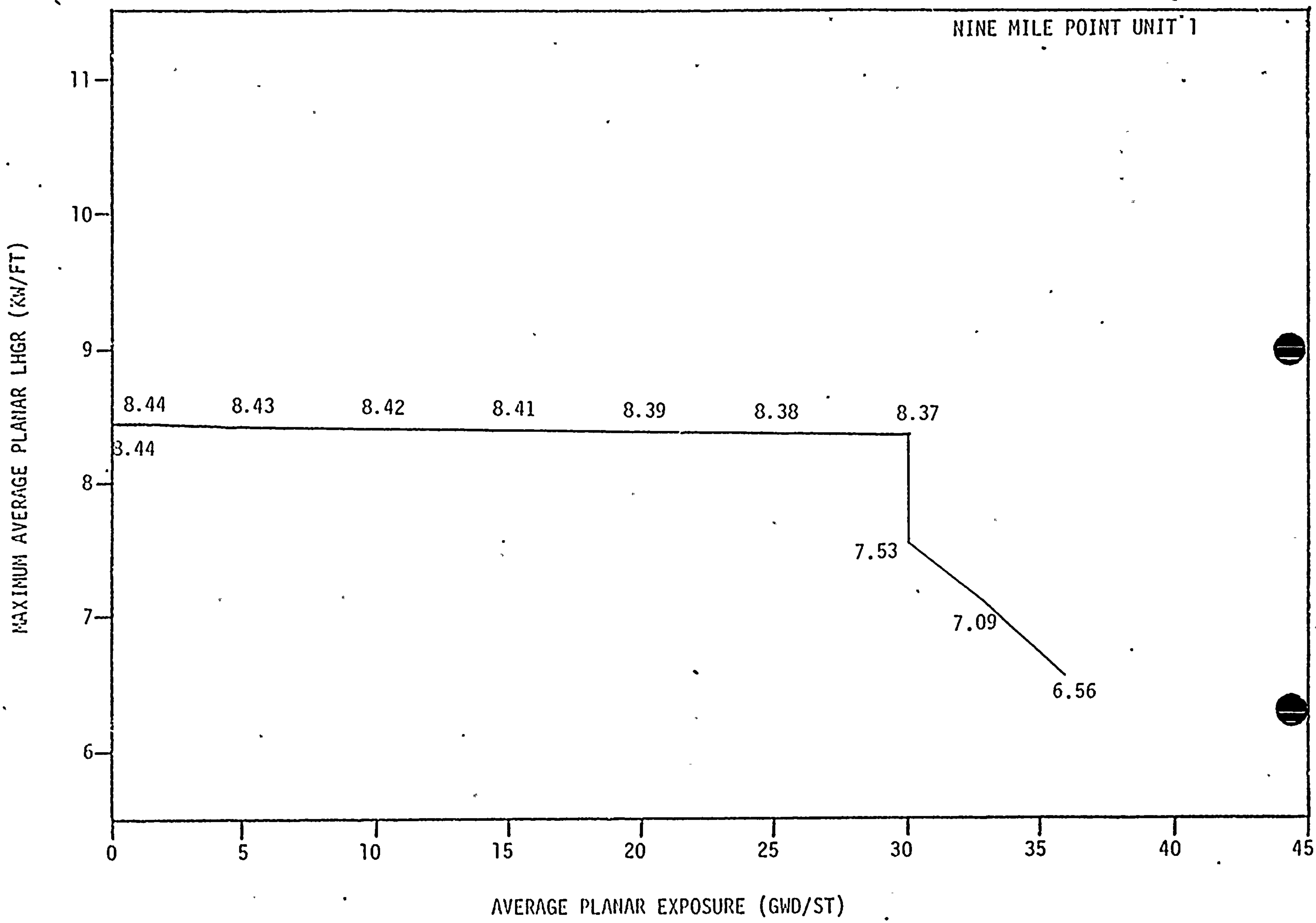

Figure 3.1.7d Maximum Allowable Average Planner LHGR Applicable to P8DNB277 and Future Reload Fuel as described in Reference 8.

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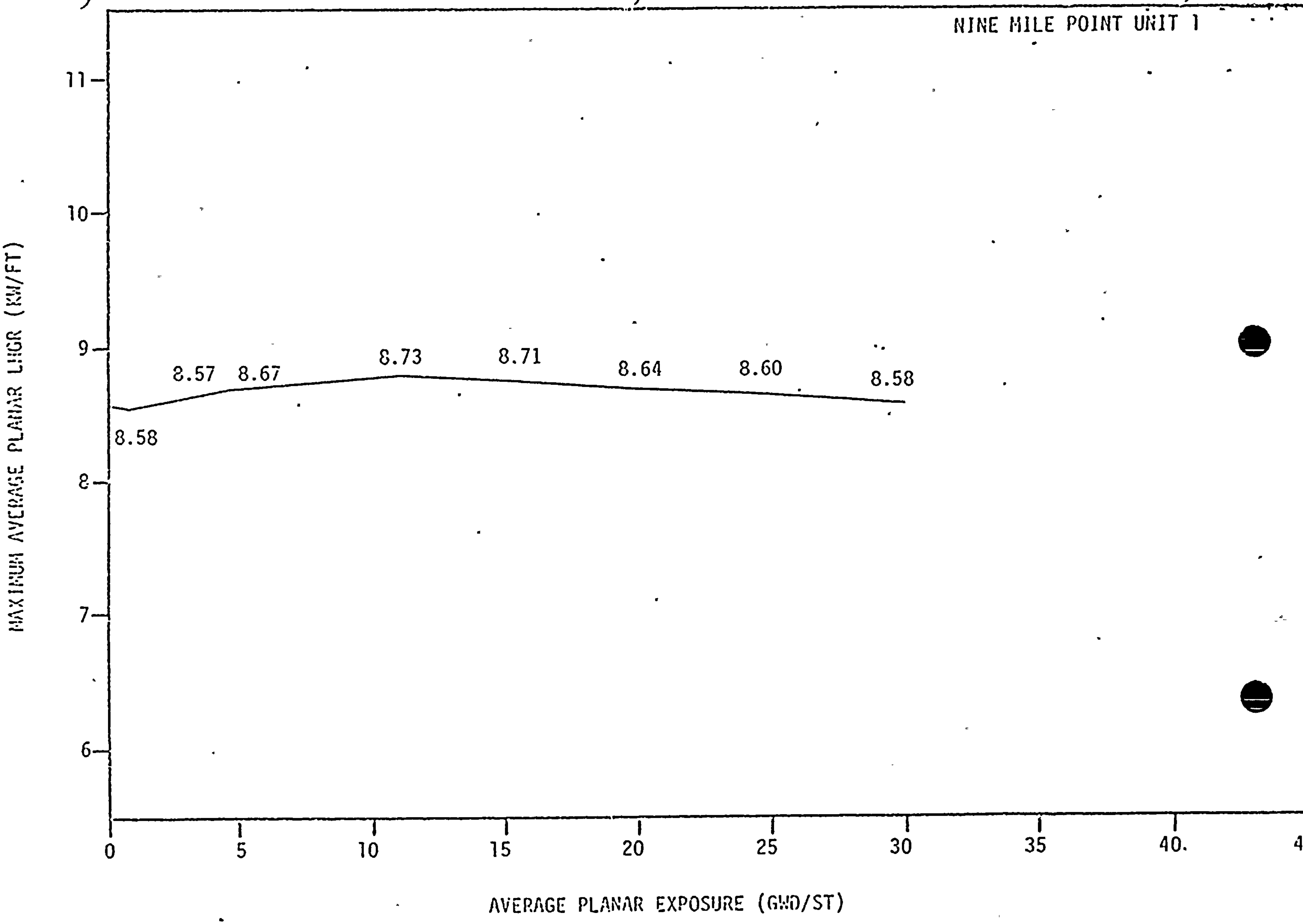

Figure 3.1.7e MAXIMUM ALLOWABLE AVERAGE PLANAR LHGR APPLICABLE TO 8DB252 FUEL AS DESCRIBED IN REFERENCE 8.

Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit 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.

Analysis has been performed (Reference 7) which shows for isolation of 1 loop, operation limited to 98% of the limiting APLHGR shown in Figure 3.1.7 conservatively assures compliance with 10CFR50, Appendix K.

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 employ 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.

Figure 3.1.7-1 is used for calculating MCPR during operation at other than rated conditions. 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).

Power/Flow Relationship

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] 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.

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 (USA 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.

b. The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR is maintained greater than the SLCPR. The trip logic for this function is 1 out of n; e.g., any trip on one of the eight APRM's, eight IRM's or four SRM's will result in a rod block. The minimum instrument channel requirements provide sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the rod block may be reduced by one for a short period of time to allow maintenance, testing, or calibration. This time period is only ∿3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM rod block trip is flow biased and prevents a significant reduction in MCPR especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than the SLCPR.

The APRM rod block also provides local protection of the core; i.e., the prevention of critical heat flux in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worst case single control rod withdrawal error has been analyzed and the results show that with the specified trip settings rod withdrawal is blocked before the MCPR reaches the SLCPR, thus allowing adequate margin. Below ∿60% power the worst case withdrawal of a single control rod results in a MCPR > SLCPR without rod block action, thus below this level it is not required.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCPR approaches the SLCPR.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and the control rod motion is prevented. The downscale rod blocks are set at 5 percent of full scale for IRM and 2 percent of full scale for APRM (APRM signal is generated by averaging the output signals from eight LPRM flux monitors).

6.0 ADMINISTRATIVE CONTROLS

6.1 Responsibility

6.1.1 The General Superintendent for Nuclear Generation shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 Organization

Offsite

6.2.1 The offsite organization for facility management and technical support shall be as shown on Figure 6.2-1.

Facility Staff

6.2.2 The Facility organization shall be as shown on Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. During reactor operation this licensed operator shall be present at the controls of the facility.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

e. A Senior Licensed Operator shall be responsible for all movement of new and irradiated fuel within the site boundary. A Licensed Operator will be required to manipulate the controls of all fuel moving equipment except the reactor building crane. All fuel movements by the reactor building crane except new fuel movements from receipt through dry storage shall be under the direct supervision of a Licensed Operator. All fuel moves within the core shall be directly monitored by a member of the reactor analyst group.

Effective until the end of fuel Cycle 6.

6.0 ADMINISTRATIVE CONTROLS

6.1 Responsibility

6.1.1 The General Superintendent for Nuclear Generation shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 Organization

Offsite

6.2.1 The offsite organization for facility management and technical support shall be as shown on Figure 6.2-1.

Facility Staff

6.2.2 The facility organization shall be as shown on Figure 6.2-2 and:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. During reactor operation, this licensed operator shall be present at the controls of the facility.

c. At least two licensed Operators shall be present in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

e. A licensed Senior Reactor Operator shall be responsible for all movement of new and irradiated fuel within the site boundary. All core alterations shall be directly supervised by a licensed senior reactor operator who has no other concurrent responsibilities during this operation. A Licensed Operator will be required to manipulate the controls of all fuel handling equipment except movement of new fuel from receipt through dry storage. All fuel moves within the core shall be directly monitored by a member of the reactor analyst group.

[a] Effective for fuel cycle 7 and all refuelings thereafter.

Table 6.2-1

MINIMUM SHIFT CREW COMPOSITION [1]

License	Normal Operation	Shutdown Condition	Operation W/O Process Computer [3]	Reactor Startups [4]
Senior Operator	1	1	1	1
Operator	2	1	2	3
Unlicensed [2]	2	1	3	2

Notes:

(1) At any one time more licensed or unlicensed operating people could be present for maintenance, repairs, fuel outages, etc.

(2) Those operating personnel not holding an "Operator" or "Senior Operator" License.

(3) For operation longer than eight hours without process computer.

(4) For reactor startups, except a scram recovery where the reason for scram is both clearly understood and corrected.

Effective until the end of fuel Cycle 6

Table 6.2-1

MINIMUM SHIFT CREW COMPOSITION[1]

License	Normal Operation	Shutdown Condition	Operation[3] W/O Process Computer	Reactor[4] Startups
Senior Operator	1	1[6]	1	1
Operator	2	1	2	3
Unlicensed[2]	2	1	3	2
Shift Technical Advisor	1	1[5]	1	1

Notes:

(1) At any one time, more licensed or unlicensed operating people could be present for maintenance, repairs, fuel outages, etc.

(2) Those operating personnel not holding an "Operating" or "Senior Operator" License.

(3) For operation longer than eight hours without process computer.

(4) For reactor startups, except a scram recovery where the reason for scram is both clearly understood and corrected.

(5) Hot shutdown condition only.

(6) An additional senior reactor operator who has no other concurrent responsibilities shall supervise all core alterations.

[a] Effective for fuel cycle 7 and all refuelings thereafter.