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TERA

Docket No. 50-220

APR 2 1979

Mr. Donald P. Dise
Vice President - Engineering
Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

Dear Mr. Dise:

The Commission has issued the enclosed Amendment No. 31 to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station, Unit 1. The amendment consists of changes to the Technical Specifications in response to your submittal dated November 21, 1978 as amended by letters dated January 2, and February 12, 1979.

The amendment revises the Technical Specifications to reflect the core reload utilizing General Electric's retrofit 8x8R fuel. For plant operations up to and including end-of-cycle (all rods out) conditions, the proposed Technical Specification changes have been found acceptable. However, as agreed to by Niagara Mohawk personnel, two licensing restrictions have been imposed for operation after the end-of-cycle during coastdown. The coastdown minimum power level is limited to 70 percent, and increasing core power via reduced feedwater heating is precluded.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,
Original Signed by
T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Const ccp

Enclosures:
1. Amendment No. 31 to License No. DPR-63
2. Safety Evaluation
3. Notice

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cc w/enclosures: See page 2

OFFICE	ORB #3	ORB #3	AD:E&P	OELD	ORB #3	
SURNAME	SSheppard	PPolk:mjf	BGrimes		TIppolito	
DATE	3/20/79	3/20/79	3/22/79	3/26/79	3/21/79	

cc: Eugene B. Thomas, Jr., Esquire
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Region II Office
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26 Federal Plaza
New York, New York 10007

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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. 31
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 November 21, 1978, as supplemented Jauary 2, and February 12, 1979, 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 paragraphs 2.C.(2) and 2.C.(3) of Facility Operating License No. DPR-63 are hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 31, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   (3) Operation beyond the end-of-cycle (all rods out condition) thermal power is limited to seventy (70) percent minimum.

   Increasing core power level via reduced feedwater heating, once operation in the coastdown mode has begun, is not allowed.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 2, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 31

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
20	20
63	63
64	64
64a	64a
64b	64b
65	65
66	66
67	67
68	68
68a	68a
69a	69a
70c	70c
237a	237a

Marginal lines indicate area of change

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 1.07 shall constitute violation of the fuel cladding integrity safety limit.

b. When the reactor pressure is less than or equal to 800 psia or core flow is less than 10% of rated, the core power shall not exceed 25% of rated thermal power.

2.1.2 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip settings on automatic protective devices related to variables on which the fuel loading safety limits have been placed.

Objective:

To provide automatic corrective action to prevent exceeding the fuel cladding safety limits.

Specification:

Fuel cladding limiting safety system settings shall be as follows:

a. The flow biased APRM scram 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.

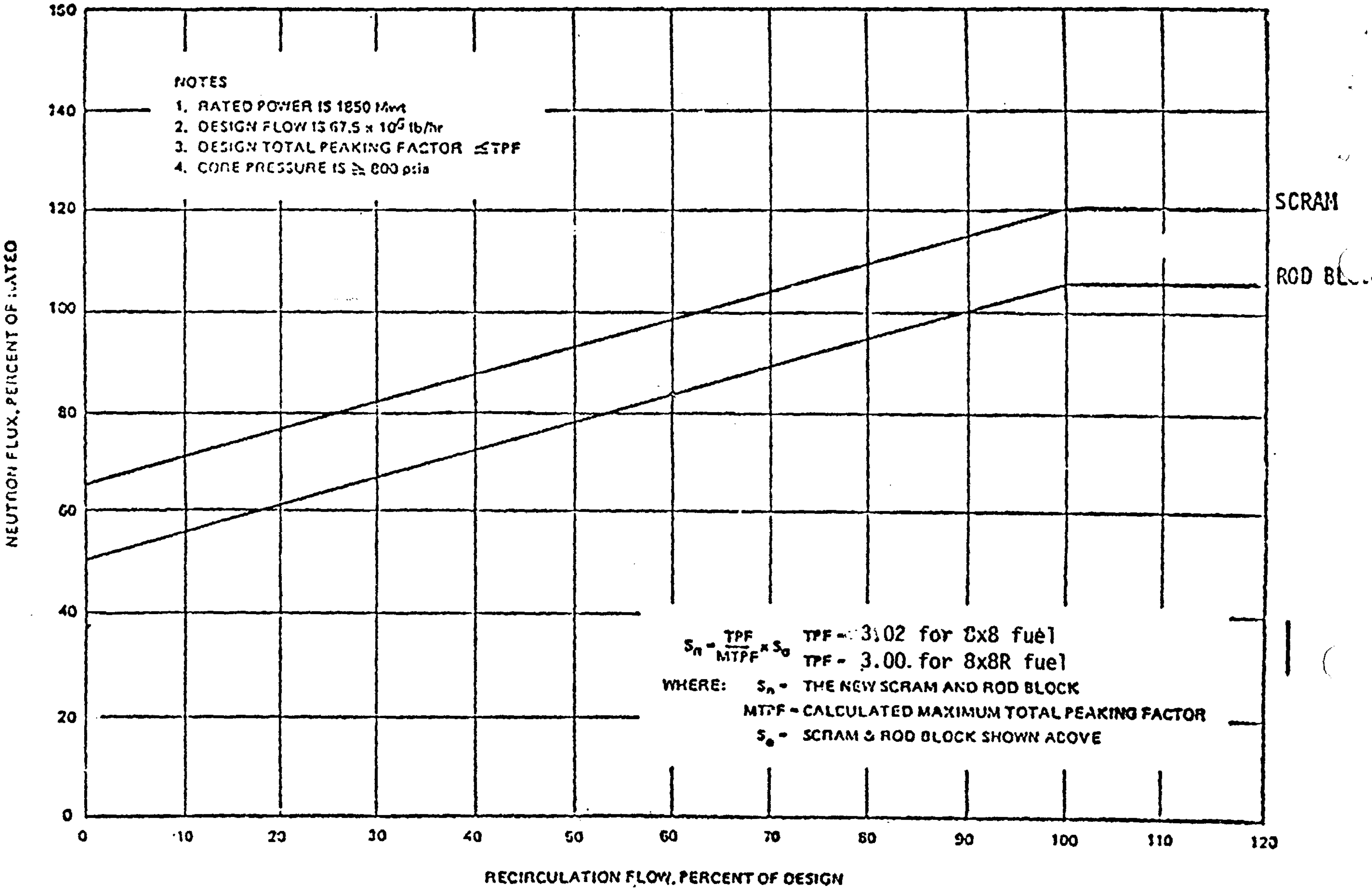

Figure 2.1.1. Flow Biased Scram and APRM Rod Block

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 1.07 MCPR > 1.07 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 safety limit (MCPR of 1.07 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 safety limit 1.07 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 MCPR = 1.07 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 (MCPR = 1.07) 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 MCPR = 1.07 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 MCPR above 1.07. 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 minimum 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 1.07. 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

(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 Boiling Water Reactor Generic Reload Fuel Application, NEDE-24011-P-A, August, 1978.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

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, and 3.1.7c. 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.

During power operation with one recirculation line isolated, the APLHGR for each fuel type as a function of average planar exposure shall not exceed 98% of limiting value shown in Figures 3.1.7a, 3.1.7b, and 3.1.7c.

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 the maximum allowable LHGR as calculated by the following equation:

$$LHGR_{max} \leq LHGR_d \left(1 - \left(\frac{\Delta P}{P}\right)_{max} \left(\frac{L}{LT}\right) \right)$$

$LHGR_d$ = Design LHGR =

13.4 kW/ft for 8x8 and 8x8R fuel

$\left(\frac{\Delta P}{P}\right)_{max}$ = Maximum power spiking penalty = 0.022 for 8x8 and 8x8R fuel

LT = Total core length - 12 ft for 8x8 fuel and 12.1033 ft for 8x8R fuel

L = Axial position above bottom of core

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 $\geq$25% rated thermal power.

LIMITING CONDITION FOR OPERATION

c. Minimum Critical Power Ratio (MCPR)

During power operation MCPR shall be $\geq$ 1.40 for 8x8 fuel and $\geq$ 1.37 for 8x8R fuel at rated power and flow. 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.

When operating the reactor with one recirculation loop isolated, core power shall be restricted to 90.5% full licensed power.

SURVEILLANCE REQUIREMENT

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.

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. Reporting Requirements

If any of the limiting values identified in Specification 3.1.7.a, b, c and d are exceeded, a Reportable Occurrence Report shall be submitted. If the corrective action is taken, as described, a thirty-day written report will meet the requirements of this Specification.

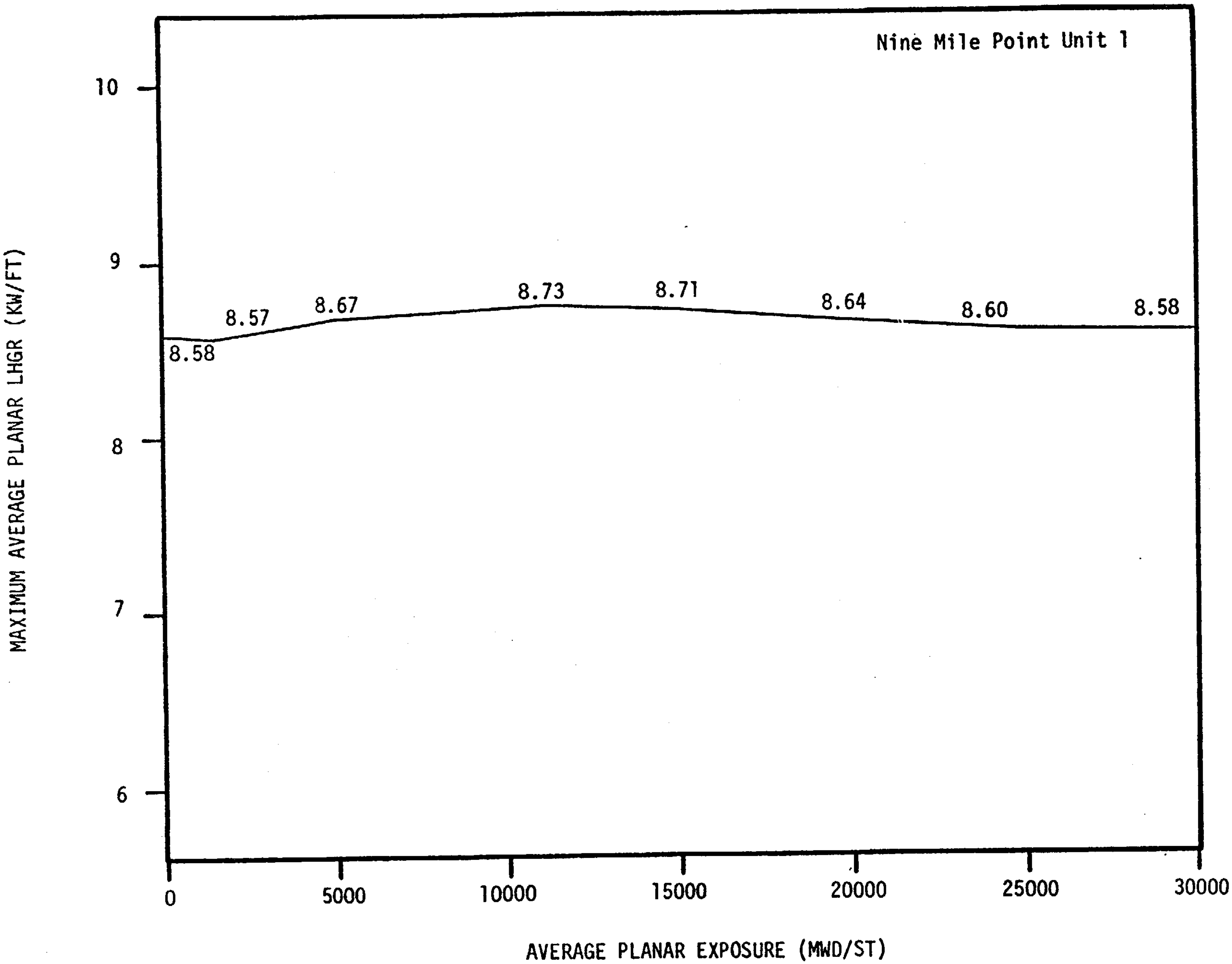

Figure 3.1.7a MAXIMUM ALLOWABLE AVERAGE PLANAR LHGR APPLICABLE TO 8DB250 and 8DB262 FUEL AS DESCRIBED IN REFERENCE 8.

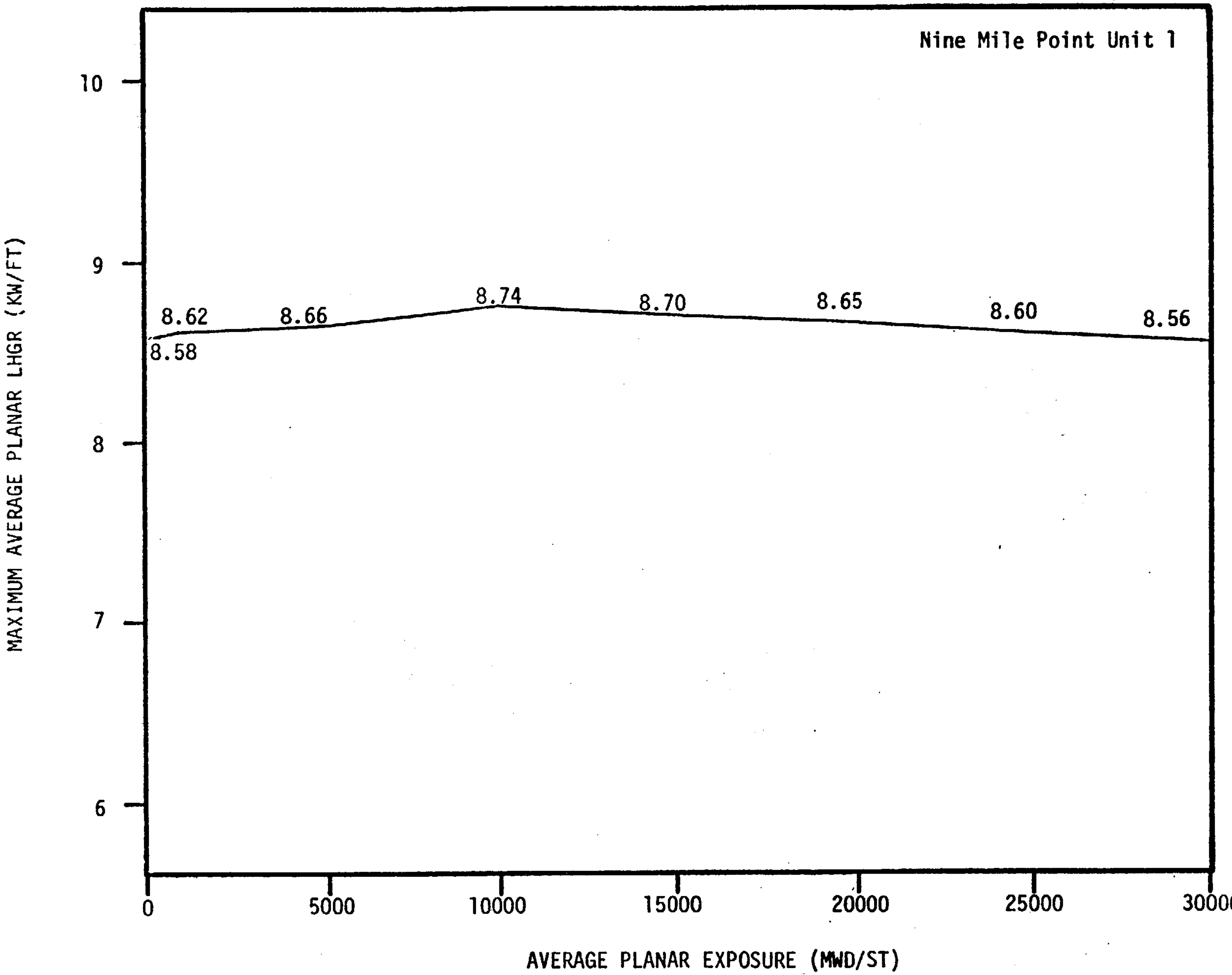

FIGURE 3.1.7b MAXIMUM ALLOWABLE AVERAGE PLANAR LHGR APPLICABLE TO 8DB274L and 8DB274H FUEL As Described in Reference 8.

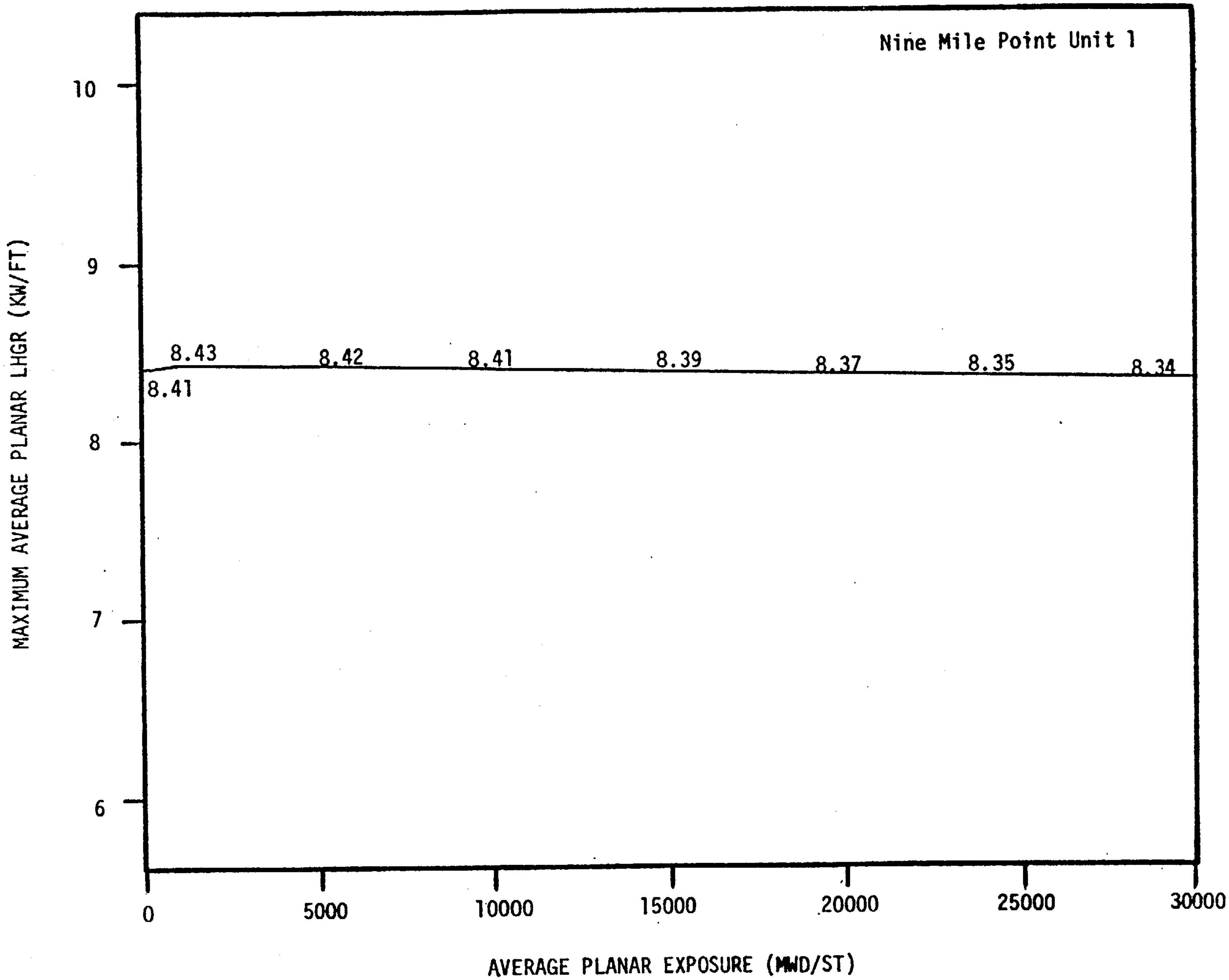

FIGURE 3.1.7c MAXIMUM ALLOWABLE AVERAGE PLANAR LHGR APPLICABLE TO 8DNB277 FUEL
As Described in Reference 8.

Amendment No. 31

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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.

b. The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR is maintained greater than 1.07. 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 for 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 1.07.

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 1.07, thus allowing adequate margin. Below ~60% power the worst case withdrawal of a single control rod results in a MCPR > 1.07 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 1.07.

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 trips are set at 5/125 of full scale for IRM and 3/125 of full scale for APRM.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 31 TO LICENSE NO. DPR-63

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-220

1.0 Introduction

By letter(1) dated November 21, 1978 and supplemented by letters(2,3) dated January 2, 1979 and February 12, 1979, the Niagara Mohawk Power Corporation (the licensee) requested amendment to the Technical Specifications appended to Operating License DPR-63 for Nine Mile Point Nuclear Station, Unit No. 1 (NMP-1). The proposed changes relate to the seventh refueling of NMP-1, involving the replacement of 76 exposed 7x7 fuel assemblies and 108 exposed 8x8 fuel assemblies with an equivalent number of fresh, two water rod, retrofit 8x8 fuel assemblies designed and fabricated by the General Electric Company. In support of this reload application for NMP-1, the licensee has submitted supplemental reload licensing documents(4,5) prepared by the General Electric Company (GE), proposed plant Technical Specification changes(6) and provided responses(3) to our request(7) for additional information on the reload application.

This reload (Reload 7) is the first for NMP-1 to utilize GE's retrofit 8x8R fuel design, although several other operating BWRs have already refueled with the new GE fuel design. Additionally, four lead retrofit 8x8 test assemblies, previously loaded into an operating reactor core, have performed satisfactorily for at least two cycles.

The descriptions of the nuclear and mechanical design of the replacement 8x8R fuel assemblies and the exposed standard 8x8 fuel assemblies, which were used in connection with the most recent NMP-1 reloads, are contained in GE's generic licensing topical report(8) for BWR reloads. Reference 8 contains a complete set of references to other GE topical reports which describe GE's BWR reload methods for the nuclear, mechanical, thermal-hydraulic, transient and accident analysis calculations. Information addressing the applicability of these methods to reload cores containing both 8x8 and 8x8R fuel is also contained in Reference 8. Portions of the plant-specific data, such as operating conditions and design parameters, used in transient and accident calculations, have also been included in the topical report.

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Our safety evaluation(9) of GE's generic reload licensing topical report concluded that the nuclear and mechanical design of the 8x8R fuel and GE's analytical methods for nuclear, thermal-hydraulic and transient and accident calculations, as applied to mixed cores containing 8x8, and 8x8R fuel, are acceptable. Our acceptance of the nuclear and mechanical design of the standard 8x8 fuel was provided in the staff's evaluation(10) of the information contained in Reference 11.

As part of our evaluation(9) of Reference 8 we found the cycle-independent input data for the reload transient and accident analyses for NMP-1 to be acceptable. The supplementary cycle-dependent information and input data are provided in Reference 4, which follows the format and content of Appendix A of Reference 8.

As a result of our generic evaluation (9) of a substantial number of safety considerations relating to the use of 8x8R fuel in mixed core loadings with 8x8 fuel, only a limited number of additional review items are included in this evaluation of Cycle 6 of NMP-1. These include the plant and cycle-specific input data and results presented in References 4 and 5, the LOCA-ECCS analysis results for the reload fuel design, and those items identified in our evaluation(9) as requiring special consideration during reload reviews.

2.0 Evaluation

2.1 Nuclear Characteristics

For Cycle 6, up to 184 fresh 8x8R fuel bundles, with a bundle average enrichment of 2.77 wt/% U-235 will be loaded into the core, replacing a like number of exposed 7x7 and 8x8 assemblies. The remainder of the 532 fuel assembly reconstituted core will consist of irradiated 8x8 fuel assemblies exposed during Cycles 4 and 5. Thus, about 35 percent of the fuel bundles are being replaced for this reload. The reference core loading for Cycle 6 will result in eighth core symmetry, which is consistent with previous cycles.

The information provided in Section 6 of References 4 and 5 indicates that the fuel temperature and void dependent characteristics of the refueled core are not significantly different from previous cycles of NMP-1. Additionally, scram effectiveness, as shown in Figures 2a, 2b and 2c of References 4 and 5, is also similar to earlier cycles. The 1.2%Δk/k calculated shutdown margin for the reconstituted core meets the Technical Specification requirement that the core be subcritical by at least 0.25%Δk/k in the most reactive operating state when the single most reactive control rod is fully withdrawn and all other rods are fully inserted. Finally, Reference 4 indicates that a boron concentration of 600 ppm in the moderator will make the reactor subcritical by 3.6%Δk at 20°C, xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria can be achieved by the Standby Liquid Control System.

2.2 Thermal Hydraulics

2.2.1 Fuel Cladding Integrity Safety Limit MCPR

As stated in Reference 9, for BWR cores which reload with GE's retrofit 8x8R fuel, the allowable minimum critical power ratio (MCPR), resulting from either core-wide or localized abnormal operational transients, is equal to 1.07. When meeting this MCPR safety limit, during a transient, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

The 1.07 safety limit minimum critical power ratio (SLMCPR) proposed by the licensee for Cycle 6 represents a .01 increase from the 1.06 SLMCPR applicable during Cycle 5. The basis for the revised safety limit is addressed in Reference 8, while our generic approval of the new limit is given in Reference 9.

2.2.2 Operating Limit MCPR

Various transient events can reduce the MCPR from its normal operating level. To assure that the fuel cladding integrity safety limit MCPR will not be violated during any abnormal operational transient, the most limiting transients have been reanalyzed for this reload by the licensee in order to determine which event results in the largest reduction in the minimum critical power ratio. These events have been conservatively analyzed for both the exposed 8x8 fuel and the reload 8x8R fuel at the most adverse cycle exposure condition.

The methods used for these calculations, including cycle-independent initial conditions and transient input parameters are described in Reference 8. Our acceptance of the values used and related transient analysis methods appears in Reference 9. Supplementary cycle-dependent initial conditions and transient input parameters used in the transient analyses appear in the tables in Sections 6 and 7 of References 4 and 5. Our evaluation of the methods used to develop these supplementary transient input values have already been addressed and appear in Reference 8. The overall transient methodology, including cycle-independent transient analysis inputs, provides an adequately conservative basis(9) for the determination of transient ΔCPRs. The transient events analyzed were: pressurization (turbine trip without bypass, and feedwater controller failure), feedwater temperature reduction (loss of 100°F feedwater heating) and local reactivity insertion (control rod withdrawal error).

The licensee reports that the most limiting event in the above categories for both the exposed 8x8 assemblies and the reload 8x8R assemblies is the control rod withdrawal error. This transient results in CPR reductions of 0.28 for the standard 8x8 assemblies and 0.30 for the retrofit 8x8 assemblies, with an Average Power Range Monitor rod block setpoint of 105%. Addition of these ΔCPRs to the 1.07 SLMCPR establishes fuel type dependent operating limit MCPRs (i.e. 1.35 for 8x8 fuel and 1.37 for 8x8R fuel) sufficient to assure that the SLMCPR will not be violated during Cycle 6 even if any of the aforementioned events were to occur.

The licensee also has considered the effects of possible fuel loading errors (FLE) on bundle CPR. The results of the licensee's FLE analysis (see Section 2.3.3 herein) shows that a somewhat higher MCPR operating limit would be required for the 8x8 assemblies in order to assure that the MCPR safety limit would not be violated in the event of the most severe FLE. In view of these results, the licensee has proposed that for Cycle 6, the 8x8 MCPR operating limit be adjusted upward from the aforementioned 1.35 to 1.40. These operating limits MCPRs (i.e., 1.40 for the 8x8 bundles and 1.37 for the 8x8R bundles) are acceptable to the staff.

2.2.3 Fuel Cladding Integrity Safety Limit LHGR

The control rod withdrawal error and fuel loading error events were analyzed by the licensee, to determine the maximum linear heat generation rates (LHGR). The results for NMP-1 Cycle 6 show that the fuel type and exposure-dependent safety limit LHGRs, shown in Table 2-3 of Reference 6 will not be violated should these events occur. Thus, fuel failure due to excessive cladding strain will be precluded should either of these events occur. These results are acceptable to the staff.

2.3 Accident Analysis

2.3.1 ECCS Appendix K Analysis

On December 27, 1974, the Atomic Energy Commission issued an Order for

Modification of License, implementing the requirements of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that prior to any license amendment authorizing any core reloading... "the licensee shall submit a re-evaluation of ECCS performance calculated in accordance with an acceptable evaluation model which conforms to the provisions of 10 CFR Part 50.46." The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation assumptions.

For Cycle 6 the licensee has re-evaluated the adequacy of NMP-1 ECCS performance in connection with the new reload fuel design, using methods previously approved by the staff. The results of these plant-specific analyses are given in Reference 4.

We have reviewed the information submitted by the licensee and conclude that NMP-1 will be in conformance with all the requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50 when operated in accordance with the MAPLHGR versus Average Planar Exposure values appearing in Section 14 of Reference 4.

2.3.2 Control Rod Drop Accident

The key plant-specific nuclear characteristics for the worst case control rod drop accident (CRDA) occurring during either cold startup or hot startup conditions are within the values used in the bounding CRDA analysis given in Reference 8. The bounding analysis shows that the peak fuel enthalpy will not exceed the 280 cal/gm design limit. Therefore, for Cycle 6 of NMP-1 the peak fuel enthalpy associated with a CRDA from either cold or hot startup conditions will also be within the 280 cal/gm design limit.

2.3.3 Fuel Loading Error

The licensee has considered the effect of postulated fuel loading errors on bundle CPR. An analysis of the most severe fuel loading errors were performed using GE's standard methods, which have previously been reviewed and approved by the staff. The results show that worst possible fuel bundle misloadings will not cause a violation of the 1.07 safety limit MCRP assuming the proposed 1.40 OLMCPR for the 8x8 fuel assemblies and 1.37 OLMCPR for the 8x8R fuel assemblies. Thus, these operating limits MCPRs will effectively preclude DNB related fuel failures caused by either fuel cladding overheating or cladding oxidation, which might otherwise occur because of a fuel loading error accident. This is acceptable to the staff.

2.4 Overpressurization Analysis

For Cycle 6 the licensee presented[4] the results of an overpressurization analysis in order to demonstrate that adequate margin exists to the ASME code allowable vessel pressure (110 percent of vessel design pressure). The transient analyzed was the closure of all main steam isolation valves with no reactor scram. The analysis was performed assuming 100 percent power, core nuclear physics parameters applicable to the end of Cycle 6, no credit for the relief function of the safety/relief valves, no reactor scram and all safety valves operative. The results of this analysis, postulated to occur during the most adverse time during the cycle, shows that the peak pressure at the vessel bottom would be 1315 psig. This provides a 60 psi margin to the 1375 psig ASME code limit.

Overpressure analyses accepted by the staff on other BWR reload applications have assumed MSIV closure with high neutron flux scram and one failed safety valve. However, the assumption of no scram for the overpressurization analysis for Reload 7 of NMP-1 represents a conservatism which we believe more than compensates for the assumption of no failed safety valve. Thus, the staff finds the 60 psi pressure margin to the 1375 psig ASME code allowable limit to be acceptable.

2.5 Thermal-Hydraulic Stability

A thermal-hydraulic stability analysis was performed for this reload using the methods described in Reference 8. The results show that the channel hydrodynamic and reactor core decay ratios at the least stable operating state (corresponding to the intersection of the natural circulation curve and the extrapolated rod block line) are 0.46 and 0.51 respectively. These are both well below the 1.0 Ultimate Performance Limit decay ratio proposed by GE.

The staff has expressed generic concerns regarding reactor core thermal-hydraulic stability at the least stable reactor condition. This condition could be reached during an operational transient from higher power if the plant were to sustain a trip of both recirculation pumps without a reactor trip. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as reload fuel designs change. The staff concerns relate to both the consequences of operating at a decay ratio of 1.0 and the capability of the analytical methods to accurately predict decay ratios. GE is addressing these staff concerns through meetings, topical reports and a stability test program. It is expected that the test results and data analysis, as presented in a final test report, will aid considerably in resolving the staff concerns.

Although we have not yet arrived at a final generic evaluation of GE's BWR stability methods and design criteria, in view of the relatively low decay ratios calculated for this reload together with the methods qualification information submitted by GE to date, we find the stability margins for Cycle 6 of NMP-1 to be acceptable.

2.6 Pressure Margin to Safety Valve Actuation

GE currently recommends[8] that for the most severe abnormal operational transient, a 25 psi margin be maintained to the lowest safety valve setpoint. The purposes of this recommendation is to prevent discharge of steam directly to the drywell, which occurs whenever the safety valves lift. This situation can be avoided if the relief valves (which discharge via piping to an underwater position in the torus) can accommodate all of the necessary excess steam flow.

For NMP-1 the worst pressurization transient is a turbine trip with bypass failure occuring at end-of-cycle. Analysis results[4] provided by the licensee, using the methods described in Reference 8, indicate that because of degrading scram effectiveness power reductions are necessary near and at end-of-cycle in order to maintain a 25 psi pressure margin. However, these initial calculations[4] incorporated conservative nuclear data which resulted in excessive end-of-cycle core power deratings. Accordingly the licensee performed a reanalysis[5] based on updated core nuclear characteristics. The results of these analyses show that a 25 psi margin is available for full power until 1500 Mwd/t prior to EOC-6. However, power limitations of 98% at EOC6-1000 Mwd/t and 95% at EOC6 are required to assure a 25 psi margin. Beginning with the first of the aforementioned exposure points, a power coast-down will be effected until the next lower power level is achieved by fixing control rod position at the start of the exposure interval. Once power falls off to the next lower power level limit, power will be maintained at that value by normal rod motion until the next exposure point is attained. This procedure will then be repeated for the second derate exposure interval.

We find the proposed power coastdown procedure, as described above, to be an acceptable method for assuring the availability of a 25 psi margin to the lowest safety valve setpoint during Cycle 6.

2.7 Power Coastdown Beyond End-of-Cycle

The licensee states[12] that operation beyond the end-of-cycle all rods out condition, in a thermal power coastdown mode, is allowable via reference to the reload topical report[8]. Although our evaluation[9] of the reload topical found the report to be acceptable for reference,we did not specifically include power coastdown operation beyond the end-of-cycle in our review. Accordingly, we do not consider the subject to have been completely addressed generically and cannot find operation in this mode acceptable on a referenced basis.

In response[3] to our request for additional information[7] on this subject, the licensee referenced power coastdown safety analyses[13,14] submitted in connection with similar requests for other operating BWRs. The referenced analyses are for particular BWRs in specific reload cycle core configurations and therefore are not explicitly applicable to Cycle 6 of NMP-1. The referenced analyses show that transient consequences regarding ΔCPR and overpressurization become less severe beyond end-of-cycle. Thus for the same operating limits, margins to core and reactor coolant pressure boundary safety limits increase for burnups beyond the end-of-cycle all rods out condition. The improved transient behavior is predominantly due to the dominant beneficial effect of reduced gross core power level in coastdown operation more than setting the secondary adverse effect of degraded scram reactivity. The analysis assumes a linear rate of power decrease with exposure, which is conservative, since actual thermal power will decrease more rapidly in an exponential manner.

As previously stated, the referenced analyses are not specifically applicable to this plant and cycle. However, we agree with the licensee's argument that the overall trend will be the same for NMP-1 during Cycle 6. Our agreement is restricted to a terminal power level of 70 percent, however. We are confident that down to 70 percent, the scram reactivity insertion rate will not be degraded sufficiently to cause a transient more severe than that of end of cycle. On the above basis we find power coastdown operation, as restricted in a license condition to not less than 70 percent power, to be acceptable. For power coastdown operations to power levels lower than 70 percent, we have requested that cycle and plant-specific analyses or other appropriate justification be provided.

Additionally, neither the current nor proposed Technical Specifications preclude increasing core power level via reduced feedwater heating once operation in the coastdown mode has begun. Such operation, although not planned at this time by the licensee, would negate the assumptions in the referenced analysis as well as the arguments and possibly the conclusions stated above. Accordingly, we require adequate assurance, in the form of a license condition, that feedwater heating capability not be reduced from the normal end-of-cycle operating configuration in order to increase reactor power once into the thermal power coastdown mode.

We have discussed these restrictions with the licensee and he has agreed to these conditions.

3.0 Physics Startup Testing

Several of the key reload safety analysis inputs and results can be assured via preoperational testing. In order to provide this assurance the licensee will perform a series of physics startup tests, which are described in Reference 3. Based on our review, this program is acceptable. A written report, describing the results of the physics startup tests, will also be provided by the licensee within 90 days of startup which is also acceptable.

4.0 Technical Specification Changes

The proposed technical specification changes(5) include a revised fuel cladding integrity safety limit MCPR, revised exposure-dependent operating limit minimum critical power ratios (MCPR) for each fuel type, addition of a MAPLHGR vs. average planar exposure curve and addition of a design maximum total peaking factor for the reload 8x8R fuel assemblies.

The revised 1.07 safety limit MCPR results in a 0.01 increase from the 1.06 safety limit MCPR (SLMCPR) used during Cycle 5. Based on our generic review(7), we find the use of a 1.07 SLMCPR for NMP-1 during Cycle 6 to be acceptable. Also, based on the discussions appearing in Section 2.2.2 herein, the staff finds the proposed operating limit MCPRs to be consistent with and adequately supported by the Reload 7 safety analyses.

The proposed 8x8R design maximum total peaking factor of 3.00 used in connection with the APRM Flux Scram and APRM Rod Block Trip Settings has been reviewed and found to be acceptable. Additionally, we find the proposed MAPLHGR vs average planar exposure curves for the 8x8R fuel assemblies to be adequate to assure conformance with the requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.

Finally, the current NMP-1 Technical Specifications require that the reactor be brought to the Cold Shutdown condition within 36 hours if any core related thermal parameter,(i.e. APLHGR, LHGR, MCPR or power/flow relationship) which is in violation of its respective operating limit, cannot be returned to within the prescribed limit within two (2) hours. The licensee states, however, that based on previous experience a core power reduction of 10 percent or less is sufficient in most cases to return the parameter to within prescribed limits. Although the violation would be corrected the current technical specifications would require that reactor power reductions be continued and Cold Shutdown conditions achieved. The proposed technical specifications would require instead that reactor power reductions at a rate not less than 10 percent per hour be initiated if all core related thermal parameters cannot be returned to within prescribed limits within two (2) hours.

Violation of any of the aforementioned core thermal operating limits will not in and of itself cause a degradation of fuel integrity which would necessitate a reactor shutdown and cooldown. We believe that the revised requirements provide for a level of operator action which is commensurate with the safety significance of the observed condition. Accordingly we find the proposed changes to be acceptable.

5.0 Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

6.0 Conclusion

We have concluded that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 2, 1979

7.0 References

1. Letter to the Director, NRR, from LeBoeuf, Lamb, Leiby and McRae (Counsel for Niagara Mohawk Power Corporation) dated November 21, 1978.

2. Niagara Mohawk Power Corporation letter (Dise) to USNRC (Ippolito) dated January 2, 1979.

3. Niagara Mohawk Power Corporation letter (Schneider) to NRC (Ippolito) dated February 12, 1979.

4. Supplemental Reload Licensing Submittal for Nine Mile Point Nuclear Power Station Unit 1 Reload No. 7, NEDO-24155, November 1978.

5. Supplemental Reload Licensing Submittal for Nine Mile Point Nuclear Power Station Unit 1, Reload No. 7 Reanalysis Supplement, NEDO-24155-1, December 1978.

6. Proposed Changes to Technical Specifications (Appendix A) appearing as Attachment A to the Letter to the Director, NRR, from LeBoeuf, Lamb, Leiby and McRae, dated November 21, 1978.

7. USNRC letter (Ippolito) to Niagara Mohawk Power Corporation (Dise) dated January 19, 1979.

8. "Generic Reload Fuel Application,"NEDE-24011-P-A, May 1977.

9. USNRC letter (Eisenhut) to General Electric (Gridley), dated May 12, 1978, transmitting "Safety Evaluation for the General Electric Topical Report,'Generic Reload Fuel Application,'(NEDO-24011-P)."

10. "Status Report on the Licensing Topical Report General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel," NEDO-20360, Revision 1 and Supplement 1; by the Division of Technical Review, Office of Nuclear Reactor Regulation, USNRC April 1975.

11. "General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel," NEDO-20360 Revision 1, Supplement 4, April 1, 1976.

12. Attachment B to the letter to the Director, ONRR from LeBoeuf, Lamb, Leiby and McRae, dated November 21, 1978.

13. R. L. Bolger (CECO) letter to B. C. Rusche (NRC), "Quad-Cities Station Unit 2 Proposed Amendment to Facility License No. DPR-30, Docekt No. 50-265," dated June 11, 1976.

14. R. L. Bolger (CECO) letter to E. G. Case (NRC), "Dresden Station Unit 2 Proposed Amendment to Facility Operating License No. DPR-19 to Permit Power Coastdown from 70% Power to 40% Power, NRC Docket No. 50-237," dated June 6, 1977.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-220

NIAGARA MOHAWK POWER CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 31 to Facility Operating License No. DPR-63 to Niagara Mohawk Power Corporation (the licensee) which revised the license and Technical Specifications for operation of the Nine Mile Point Nuclear Station, Unit No. 1 (the facility) located in Oswego County, New York. The amendment is effective as of its date of issuance.

The amendment revises the Technical Specifications to reflect the core reload utilizing General Electric's retrofit 8x8R fuel.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated November 21, 1978, as supplemented January 2 and February 12, 1979, (2) Amendment No. 31 to License No. DPR-63, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Oswego County Office Building, 46 E. Bridge Street, Oswego, New York 13126. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 2nd day of April 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors