JUN 2 7 1977

Docket No. 50-220

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Niagara Mohawk Power Corporation
ATTN: Mr. Gerald K. Rhode
Vice President - Engineering
300 Erie Boulevard West
Syracuse, New York 13202

Gentlemen:

The Commission has issued the enclosed Amendment No. 16 to Facility License No. DPR-63 for Unit No. 1 of the Nine Mile Point Nuclear Station. This amendment consists of changes to the Technical Specifications and License Restrictions and is in response to your requests dated December 7, 1976 (supplemented by letter dated March 14, 1977) and March 24, 1977.

The amendment will modify the Technical Specifications to permit operation of the facility with 160 General Electric (GE) 8 x 8 reload fuel bundles and to require the use of the rod worth minimizer for power levels below 20% of rated thermal power.

The amendment also modifies the License Restriction that defines the power operation near end-of-cycle. The restriction was discussed and approved by your staff.

The staff has also reviewed Licensee Event Report 77-15 dated April 13, 1977, and has arrived at a favorable conclusion to resume operation. The evaluation is included in the enclosed Safety Evaluation.

Copies of the related Safety Evaluation and the Federal Register Notice also are enclosed.

Sincerely,

Original signed by

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 16 to License DPR-63
2. Safety Evaluation
3. Federal Register Notice

*SEE PREVIOUS YELLOW FOR CONCURRENCES

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DATE	6/ /77	6/ 21 /77	6/ 22 /77	6/ 27 /77		

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Sincerely,

George Lear, Chief
Operating Reactors Branch #3
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cc w/encls:
See next page

OFFICE➤	ORB#3	ORB#3	OELD	ORB#3	
SURNAME➤	CParrish	SNowicki:acr	~~[illegible]~~	GLear	
DATE➤	5/ 26 /77	5/ 26 /77	~~5/ 27~~ /77	5/ /77	

NRC FORM 318 (9-76) NRCM 0240

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NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 16
License No. DPR-63

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

   A. The applications for amendment by Niagara Mohawk Power Corporation (the licensee) dated December 7, 1976, (supplemented by letter dated March 14, 1977) and March 24, 1977, comply 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 applications, 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; and

   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, respectively, to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 16 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   (3) Restrictions

   Beyond the point in the Cycle 5 fuel cycle at which the reactivity reduction rate during a scram is less than that of the curve marked 2.0 GWD/T before EOC 5 in Figure 6-8 of "General Electric BWR Reload 6 Licensing Submittal for Nine Mile Point Unit 1", NEDO-21466 dated November 1976, operation of the reactor shall not exceed a core thermal power of 1740 megawatts (94% of rated).

   Beyond the point in the Cycle 5 fuel cycle at which the reactivity reduction rate during a scram is less than that of the curve marked 1.0 GWD/T before EOC 5 in Figure 6-8 of "General Electric BWR Reload 6 Licensing Submittal for Nine Mile Point Unit 1", NEDO-21466 dated November 1976, operation of the reactor shall not exceed a core thermal power of 1700 megawatts (92% of rated).

   Beyond the point in the Cycle 5 fuel cycle at which the reactivity reduction rate during a scram is less than that of the curve marked EOC in Figure 6-8 of NEDO-21466, operation of the reactor is not authorized."

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

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 27, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 16

FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contains vertical lines indicating the area of change. Add pages 64f and 69b.

Remove	Insert
29	29
35	35
37	37
63	63
64b	64b
64c	64c
64d	64d
64e	64e
—	64f
69a	69a
—	69b

LIMITING CONDITION FOR OPERATION

(b) Whenever the reactor is in the startup or run mode below 20% rated thermal power, no control rods shall be moved unless the rod worth minimizer is operable, except as noted in 4.1.1.b(3)(a)(iv), or a second independent operator or engineer verifies that the operator at the reactor console is following the control rod program. The second operator may be used as a substitute for an inoperable rod worth minimizer during a startup only if the rod worth minimizer fails after withdrawal of at least twelve control rods.

(4) Control rods shall not be withdrawn for approach to criticality unless at least three source range channels have an observed count rate equal to or greater than three counts per second.

SURVEILLANCE REQUIREMENT

(iv) The rod block function of the rod worth minimizer shall be verified by attempting to withdraw an out-of-sequence control rod beyond the block point.

(b) If the rod worth minimizer is inoperable while the reactor is in the startup or run mode below 20% rated thermal power and a second independent operator or engineer is being used he shall verify that all rod positions are correct prior to commencing withdrawal of each rod group.

(2) The rod housing support is provided to prevent control rod ejection accidents. Its design is discussed in Section VII-E.* Procedural control shall assure that the housing supports are in place for all control rods.

(3) Control rod withdrawal and insertion sequences are established to assure that the maximum in-sequence individual control rod or control rod segments which are withdrawn could not be worth enough to cause the core to be more than 0.013 delta k supercritical if they were to drop out of the core in the manner defined for the Rod Drop Accident.[3] These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow the sequences is backed up by the operation of the RWM. This 0.013 delta k limit, together with the integral rod velocity limiters and the action of the control rod drive system, limits potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy content of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in reference 1.

Recent improvements in analytical capability have allowed more refined analysis of the control rod drop accident. These techniques have been described in a topical report, two supplements and letters to the AEC.[1][2][3][4][5]. By using the analytical models described in these reports coupled with conservative or worst-case input parameters, it has been determined that for power levels less than 20% of rated power, the specified limit on in-sequence control rod or control rod segment worths will limit the peak fuel enthalpy content to less than 280 cal/gm. Above 20% power, even multiple operator errors cannot result in a peak fuel enthalpy content of 280 cal/gm should a postulated control rod drop accident occur.

The following conservative or worst-case bounding assumptions have been made in the analysis used to determine the specificed 0.013 delta k limit on in-sequence control rod or control rod segment worths. The allowable boundary conditions used in the analysis are quantified in references (4) and (5). Each core reload will be analyzed to show conformance to the limiting parameters.

*FSAR

The RWM provides automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. It serves as an independent backup of the normal withdrawal procedure followed by the operator. In the event that the RWM is out of service when required, a second independent operator or engineer can manually fulfill the operator-follower control rod pattern conformance function of the RWM. In this case, procedural control is exercised by verifying all control rod positions after the withdrawal of each group, prior to proceeding to the next group. Allowing substitution of a second independent operator or engineer in case of RWM inoperability recognizes the capability to adequately monitor proper rod sequencing in an alternate manner without unduly restricting plant operations. Above 20% power, there is no requirement that the RWM be operable since the control rod drop accident with out-of-sequence rods will result in a peak fuel energy content of less than 280 cal/gm. To assure high RWM availability, the RWM is required to be operating during a startup for the withdrawal of a significant number of control rods for any startup.

(4) The source range monitor (SRM) system performs no automatic safety function. It does provide the operator with a visual indication of neutron level which is needed for knowledgeable and efficient reactor startup at low neutron levels. The results of reactivity accients are functions of the initial neutron flux. The requirement of at least 3 cps assures that any transient at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to critical using homogeneous patterns of scattered control rods. A minimum of three operable SRM's is required as an added conservation.

c. Scram Insertion Times

The revised scram insertion times have been established as the limiting condition for operation since the postulated rod drop analysis and associated maximum in-sequence control rod worth are based on the revised scram insertion times. The specified times are based on design requirements for control rod scram at reactor pressures above 950 psig. For reactor pressures above 800 psig and below 950 psig the measured scram times may be longer. The analysis discussed in the next paragraph is still valid since the use of the revised scram insertion times would result in greater margins to safety valves lifting.

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.

Ojbective:

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 3.1.7.a, 3.1.7.b, 3.1.7.c, 3.1.7.d 3.1.7.e and 3.1.7.f. If at any time during power operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.1.7 FUEL RODS

Applicability:

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

Objective:

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

Specification:

a. Average Planar Linear Heat Generation Rate (APLHGR)

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

d. Assembly Averaged Power-Void Relationships (Cont'd)

$$\frac{(1-VF)}{PR \times FCP} \geq B$$

Where: VF = Bundle average void fraction

PR = Assembly radial power factor

FCP = Fractional core power relative to 1850 MWt

B = Power-Void Limit (limiting values of "B" shall be specified for each fuel type in the core)

The limiting values of "B" for each fuel type are shown in the Figures below:

Fuel Type(s)	"B"
Type 1 - initial core	Figure 3.1.7.aa
Type 2 - reload 1	Figure 3.1.7.bb
Type 3 - reload 2	Figure 3.1.7.cc
Type 4 - reload 3	Figure 3.1.7.dd
Type 5 - 8D250	Figure 3.1.7.ee
Type 6 - 8D262	Figure 3.1.7.eee
Type 7 - 8D274L & 8D274H	Figure 3.1.7.ff

e. Reporting Requirements

If any of the limiting values identified in Specifications 3.1.7.a, b, or c 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 3.1.7.aa

Type 1 - Initial Core Fuel

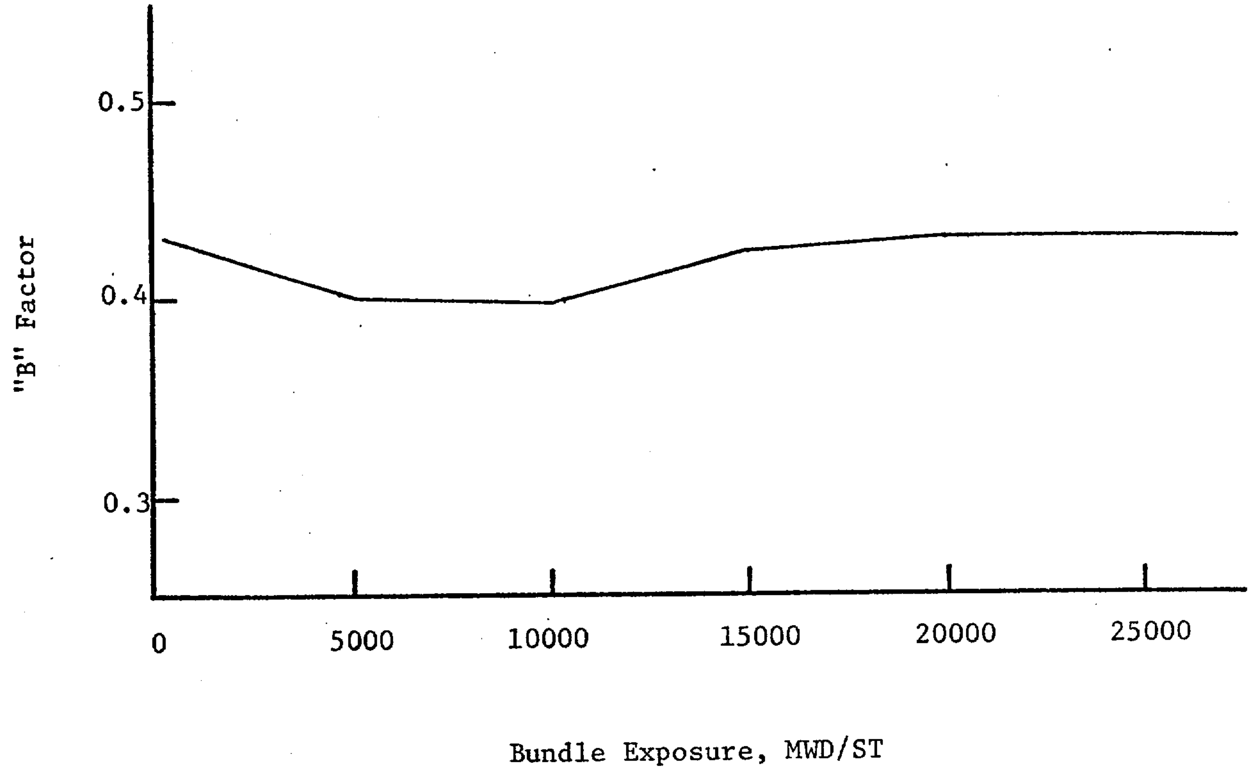


Figure 3.1.7.bb

Type 2 - Reload 1 Fuel

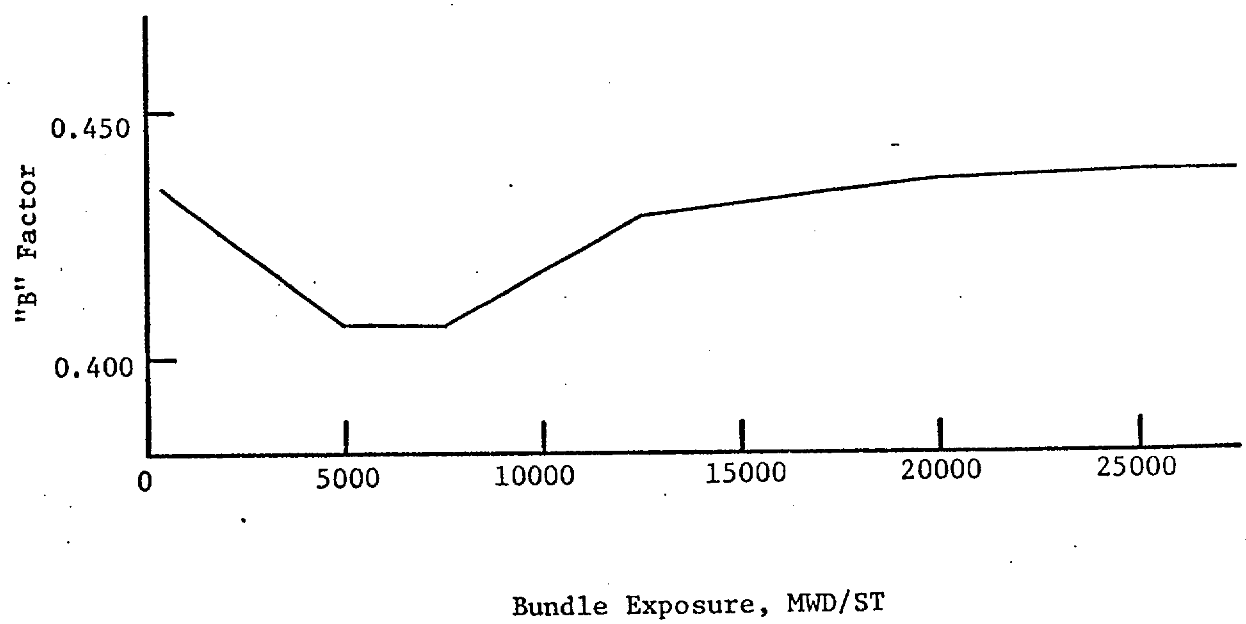


Figure 3.1.7. cc

Type 3 - Reload 2 Fuel

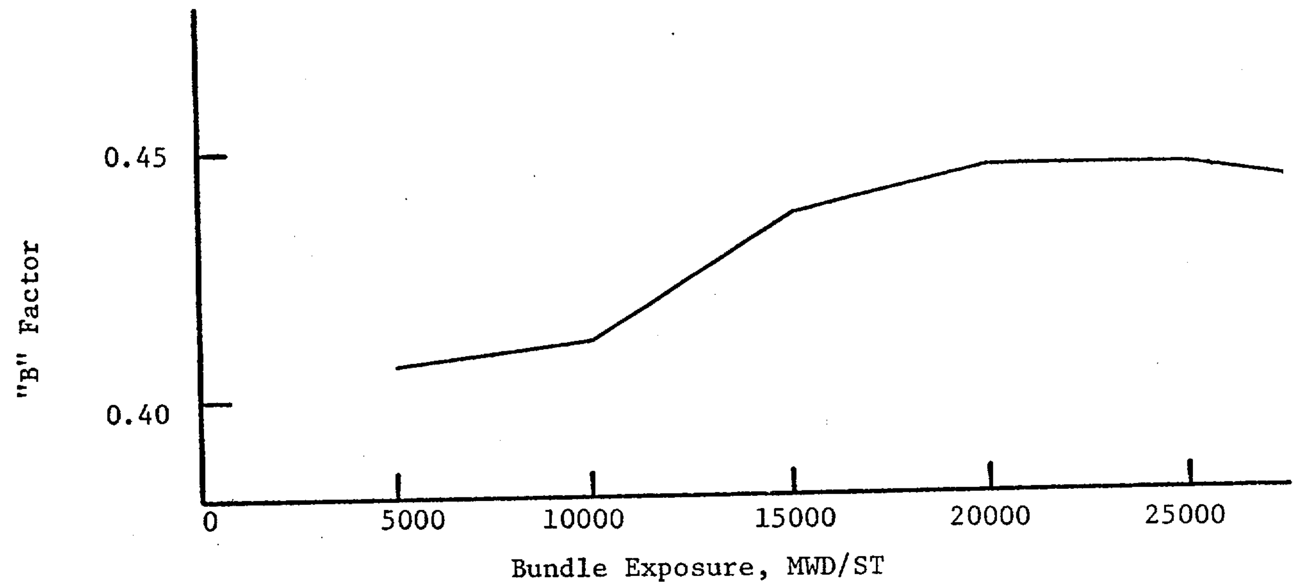


Figure 3.1.7. dd

Type 4 - Reload 3 Fuel

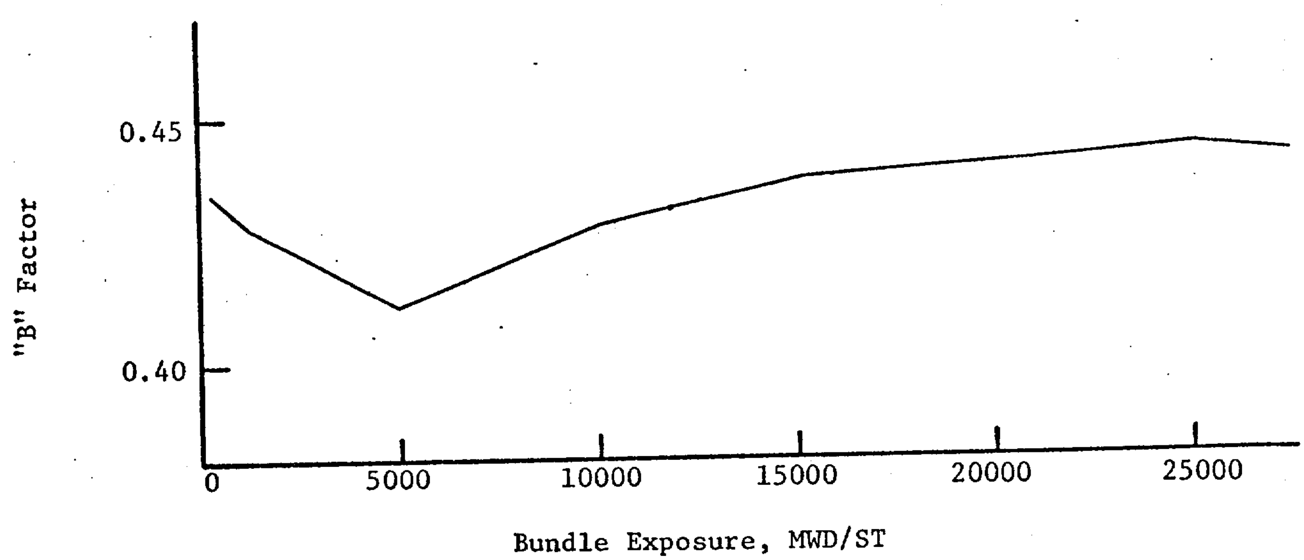


Figure 3.1.7.ee

Type 5 - 8D250

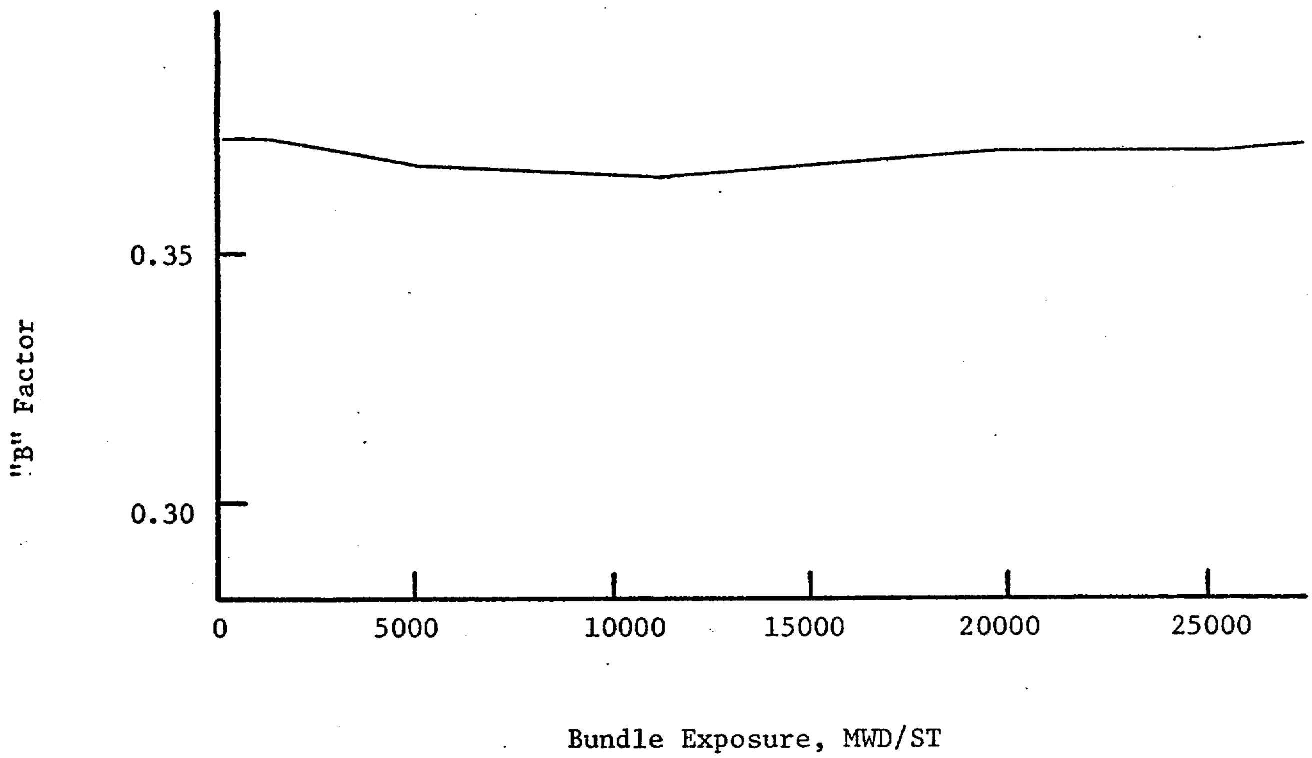


Figure 3.1.7.eee

Type 6 - 8D262

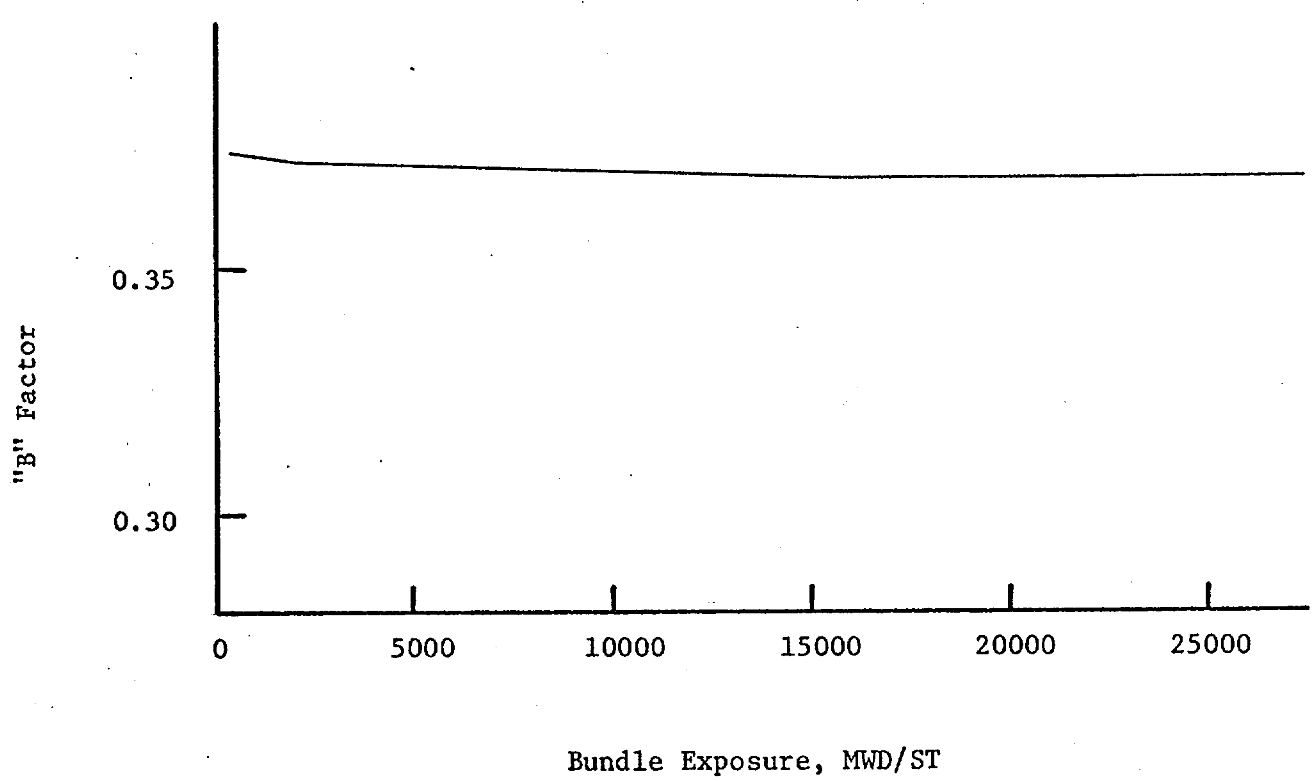
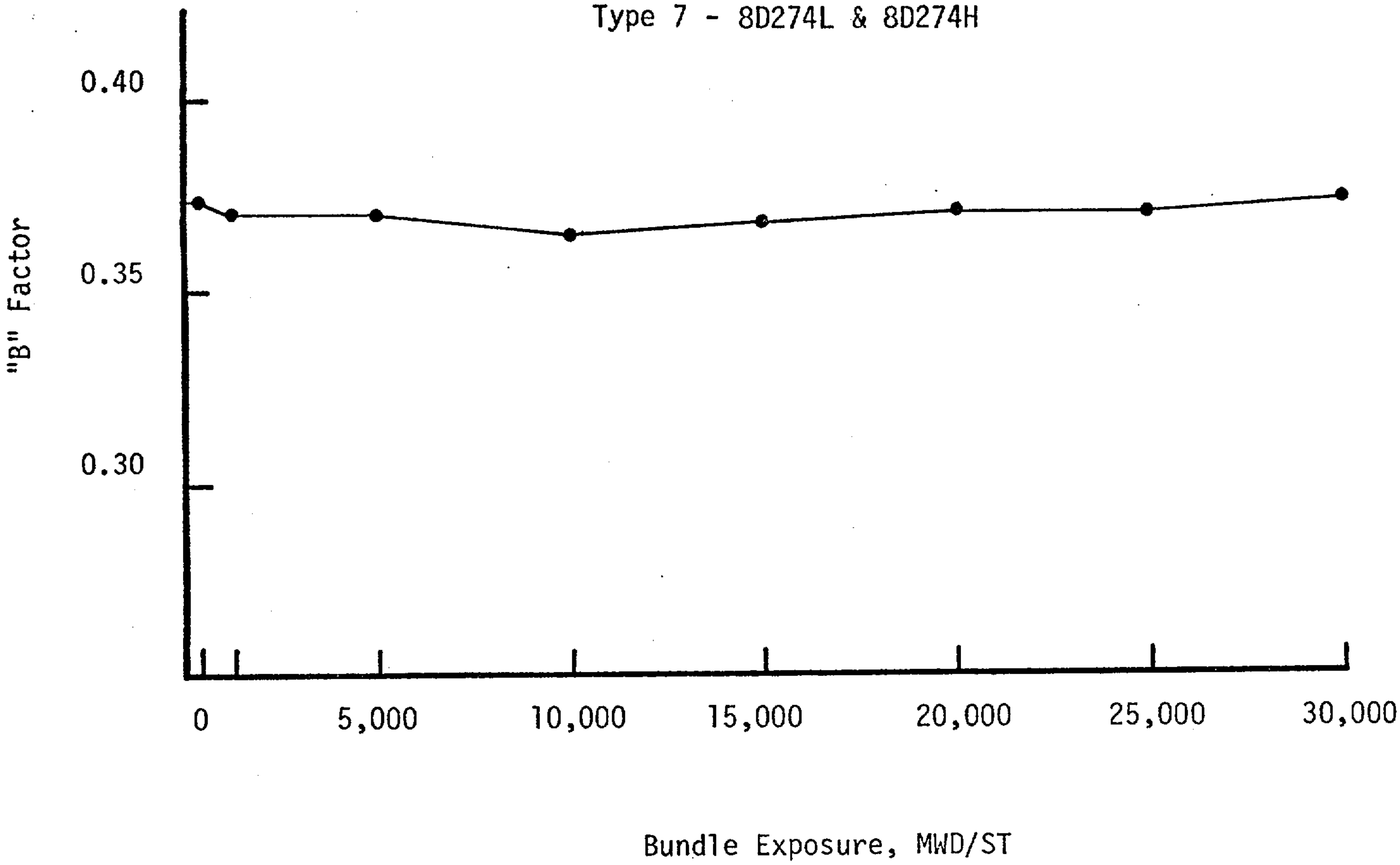


Figure 3.1.7ff

Type 7 - 8D274L & 8D274H



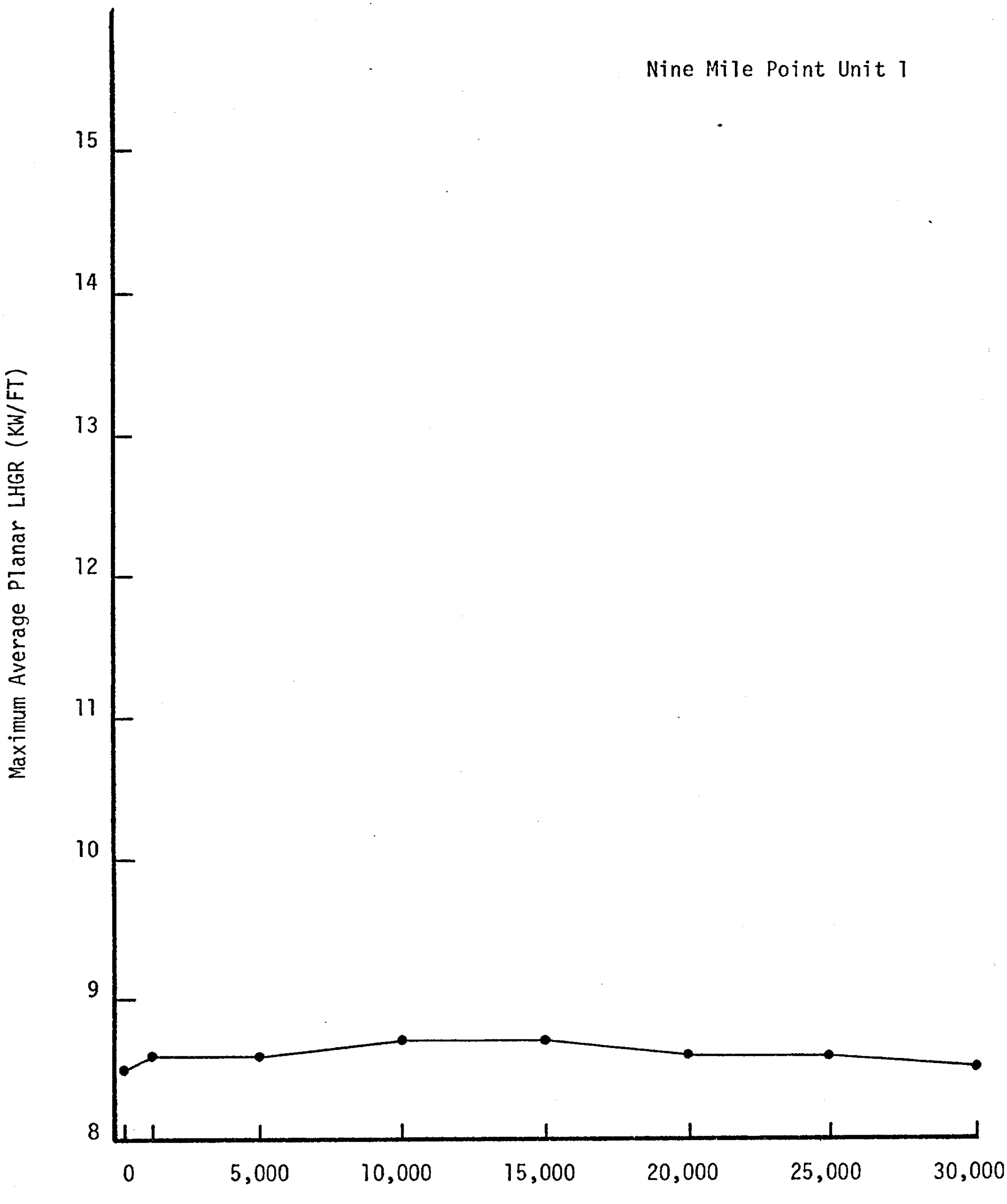


Figure 3.1.7.f Maximum Allowable Average Planar LHGR
Applicable to Type 7 Fuel

Amendment No. 16

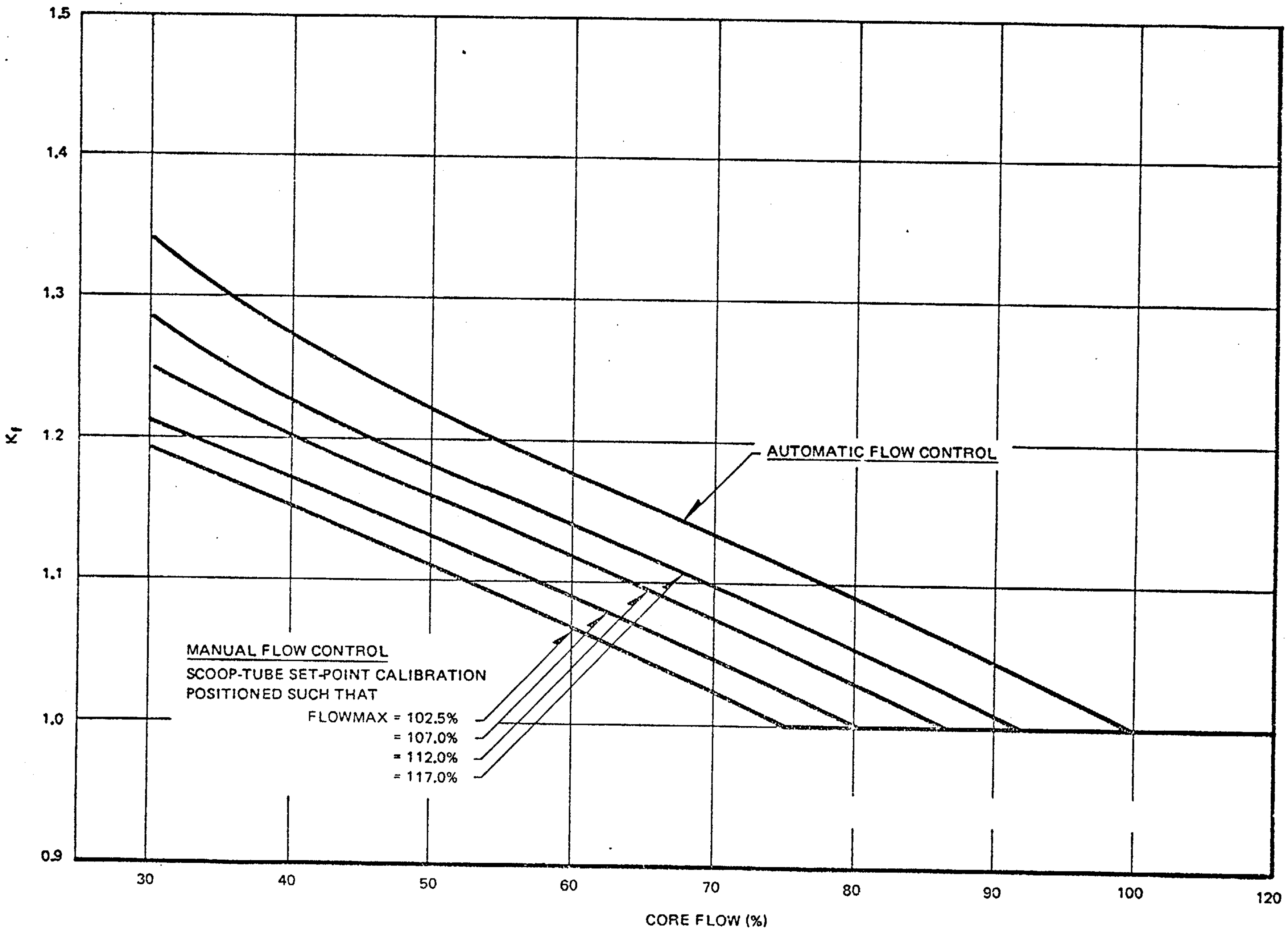


Figure 3.1.7-1 K_f Factor

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 16 TO FACILITY OPERATING LICENSE NO. DPR-63

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT UNIT NO. 1

DOCKET NO. 50-220

1.0 Introduction

By letters dated December 7, 1976 (supplemented by a letter dated March 14, 1977) and March 24, 1977, Niagara Mohawk Power Corporation (NMPC) requested an amendment to Facility Operating License No. DPR-63. The amendment would modify the Technical Specifications and the License Restrictions for the Nine Mile Point Unit No. 1 (NMP-1) to permit operation of the facility with as many as 160 General Electric (GE) 8 x 8 reload fuel bundles with an average enrichment of 2.74 weight % U-235.

2.0 Background

a. Reload and Technical Specifications

The licensee has proposed to reload the Nine Mile Point Unit No. 1 with 160 GE 8 x 8 fuel bundles with an average enrichment of 2.74 wt % U-235. The licensee has also proposed to change the Technical Specification to require that the rod worth minimizer (RWM) be operational during control rod motion for powers below 20% of rated thermal power. The current Technical Specifications require use of the RWM only up to 10% of rated thermal power. The licensee has also requested that the License Restrictions be modified to be consistent with Cycle 5 scram reactivity characteristics.

The documentation submitted in support of the proposed reload consisted of the "GE BWR Reload 6" licensing submittal for NMP-1 for the 8 x 8 fuel[1] with proposed Technical Specification changes[2, 11] supplemented with responses to NRC questions[3, 4] and referencing certain sections of a generic GE topical report on 8 x 8 reloads[5] On the bases stated in this safety evaluation, we conclude that those analyses, the Technical Specification changes and License Restriction changes are acceptable.

b. Reactor Water Level During Maintenance

During the refueling outage, in preparation for Cycle 5 operation, Feedwater Nozzle Inspection and Maintenance work was performed in the reactor vessel. Technical Specifications allow the reactor water level to be lowered 9 feet below the normal water level during the maintenance operation. On April 4, 1977 a shutdown cooling system pump tripped. The pump trip allowed the water level to drop to approximately 9 feet 3 inches below normal water level and therefore was in violation of the safety limit stated in the Technical Specifications.

The staff has reviewed the event that allowed the water level to drop 3 inches below the safety limit stated in the Technical Specification and concluded that no danger to the public health and safety occurred during or following the event. Moreover, our analysis of the incident provides a basis for the Commission, pursuant to Section 50.36 of Title 10 of the Code of Federal Regulations, to authorize NMPC to resume operation of Nine Mile Point Unit No. 1.

3.0 Evaluation

3.1 Nuclear Characteristics

The information presented in the licensing submittal closely follows the guidelines of Appendix A of NEDO-20360[5]. This topical has been found acceptable for use for reactors containing 8 x 8 reload fuel. Up to 160 8 x 8 reload fuel bundles with an average enrichment of 2.74% by weight will be loaded throughout the core. Forty of the reload fuel bundles have high gadolinia content (8D274H) and 120 have a low gadolinia content (8D274L). The core contains a total of 532 fuel bundles. Thus, about 30 percent of the fuel bundles are being replaced for the reload.

The loading pattern consists of the 8 x 8 reload bundles with low and high gadolinia content scattered throughout the core. The data in Reference 1 indicate that the nuclear characteristics of the Reload 6 core (which is Cycle 5) are similar to Cycle 4. Thus, the total control system worth, temperature, and void dependent behavior of the Cycle 5 core will not differ significantly from those values previously reported for the NMP-1 reactor. The shutdown margin of the Cycle 5 core meets the Technical Specification requirement that the core be at least 0.0025Δk subcritical in the most reactive operating state with the most reactive rod fully withdrawn and with all the others fully inserted. For the Cycle 5 core the minimum shutdown margin is 0.0144Δk, which occurs at the beginning of the cycle.

The information presented in Reference 1 indicates that a boron concentration of 600 ppm in the moderator will bring the reactor subcritical by ≥0.03Δk at 20°C, xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria is met by the Standby Liquid Control System.

The Technical Specification requirement for the storage of fuel for NMP-1 is that the effective multiplication factor, k_{eff}, of all of the fuel when stored in the fuel storage rack be ≤.90 for normal conditions. This is achieved if the uncontrolled k_{eff} of a single fuel bundle is less than 1.30 at 65°C.[5] The highest reactive fuel, the 8x8 8D274H and 8D274L fuel bundles, at both the zero exposure and the peak reactivity point, have a maximum $k_\infty < 1.25$ and, therefore, meet the fuel storage requirement for NMP-1.

The full power scram reactivity curves used for the Cycle 5 core are shown in Figure 6.8 of Reference 1. The scram curves used in the anticipated transient analyses include a design conservatism factor of 0.8[3], which is acceptable.

The void and Doppler coefficients of reactivity for the Cycle 5 core are given in Table 6-1 of Reference 1. The void coefficient of reactivity and the Doppler coefficient were compared to the Cycle 5 values given in Table 6-1 of Reference 12, and were found to be different by only a small percentage. Appropriate values for Cycle 5 were used for the Cycle 5 re-analysis, except as noted in this SER for the overpressure analysis (Section 2.5).

Therefore, due to the findings stated above, we conclude that the nuclear characteristics and performance of the Cycle 5 core are acceptable, and that they will not differ significantly from that of the previous fuel cycle, which was acceptable.

3.2 Mechanical Design

The two types of Reload 6 fuel have the same mechanical configuration and fuel bundle enrichments as the 8D274L and the 8D274H fuel assemblies described in the 8 x 8 generic reload report (Reference 5), including the improved water rod design and the finger springs described in Section 3 of Reference 5.

The generic 8 x 8 reload report (Reference 5) has been found acceptable for use for reactors containing 8 x 8 reload fuel, when supplemented with information required by our status report on the GE generic report evaluation (Reference 6). On the basis of our review of the generic 8 x 8 reload report [5, 6] and the reload submittal,[1] we conclude that the NMP-1 Reload 6 fuel mechanical design is acceptable.

3.3 Thermal-Hydraulics

The GE generic 8 x 8 fuel reload topical report[5] and GETAB[7] methodology establishes:

(1) the fuel damage safety limit,

(2) the limiting conditions of operation such that the safety limit is not exceeded for normal operation and anticipated transients, and

(3) the limiting conditions of operation such that the initial conditions assumed in the accident analyses are satisfied.

We have evaluated the NMP-1 Cycle 5 core thermal margins based on GETAB,[7] plant specific input information provided by the licensee,[1,4] and the generic 8 x 8 reload report[5]. The staff evaluation of these margins is reported herein.

3.3.1 Fuel Cladding Integrity Safety Limit MCPR

The fuel cladding integrity safety limit MCPR for NMP-1 is 1.06. It is based on the generic GETAB statistical analysis given in Reference 5 which assures that 99.9% of the fuel rods in the core are not expected to experience boiling transition during abnormal operational transients. The uncertainties in the core and system operating parameters and the GEXL correlation (Table 4-1) of the licensee submittal[1] combined with the relative bundle power distribution in the core form the basis for the GETAB statistical determination of the safety limit (SL) MCPR. The tabulated list of uncertainties for the NMP-1 Cycle 5 core are the same or more conservative than those used in GETAB.[7]

The core selected for the generic GETAB safety limit MCPR statistical analysis in Reference 5 is a typical 251/764 core which is larger than the NMP-1 core. The generic GETAB statistical analysis results are therefore conservative for application to NMP-1 since the bundle power distribution in the larger 251/764 core analyzed in the generic SL-MCPR calculation has more high power bundles than the distribution expected during Cycle 5 of operation of the smaller NMP-1 reactor. This results in a conservative value of the MCPR which meets the 99.9% criterion. We, therefore, conclude that the proposed fuel integrity safety limit, a MCPR of 1.06, is acceptable for the NMP-1 Cycle 5 core.

3.3.2 Operating Limit MCPR

Various transient events will reduce the MCPR below the operating MCPR. To assure that the fuel cladding integrity safety limit (MCPR of 1.06) is not violated during anticipated abnormal operational transients, the most limiting transients have been analyzed to determine which one results in the largest reduction in critical power ratio (ΔMCPR). The licensee has submitted the results of analyses of those transients which produce a significant decrease in MCPR (References 1 and 3). The types of transients evaluated were overpressure (turbine trip without bypass), feedwater (f-w) temperature decrease (loss of f-w heater), and reactivity increase events (worst misloading error and uncontrolled withdrawal of the most reactive rod). The most limiting transient in these categories for the 8 x 8 and 7 x 7 fuel was the rod withdrawal error transient, resulting in a ΔMCPR of 0.32 for 8 x 8 fuel and 0.30 for 7 x 7 fuel, with the Average Power Range Monitor (APRM) rod block setpoint at 105% and assuming the most limiting unavailabilities in the APRM system. Addition of these ΔMCPR's to the safety limit MCPR (1.06) gives the minimum operating limit MCPR for each fuel type required to avoid violating of the safety limit, should this limiting transient occur. Therefore, the operating limit MCPR's are 1.38 for 8 x 8 fuel and 1.36 for 7 x 7 fuel.

The transient analyses were evaluated with scram reactivity insertion rates that include a design conservatism factor of 0.80. The initial conditions[1] used for the worst operational transient are conservative with respect to actual plant operating limits or equal to plant limits and are therefore acceptable. The initial MCPRs assumed in the transient analyses were equal to or greater than the established operating limit MCPRs, which results in conservative (large) prediction of ΔMCPR.

A GE study[7] has shown that the required operating MCPR varies with the axial and local power peaking distribution. Axial peaking in the middle or upper portion of the core results in higher required MCPR's than peaking in the lower portion of the core. In the analyses the axial power peak was assumed to be representative of beginning-of-cycle conditions and to be located at the midplane (axial peak-to-peak average of 1.40).

The R-factors, which are a function of the local power peaking assumed in the analyses, are also representative of beginning-of-cycle condition. The values used are 1.10 for 7 x 7 fuel and 1.102 for 8 x 8 fuel. During the cycle, the local peaking, and therefore the R-factor, is reduced while the peak in the axial shape moves toward the bottom of the core. Although the operating limit MCPR would be increased by approximately 1% by the reduced end-of-cycle R-factor, this is offset by the reduction in MCPR resulting from the relocation of the axial peak to below the midplane.

Conservatism was applied in the determination of the required operating limit MCPR because the assumed axial and local peaking were representative of the beginning of the fuel cycle. This is the worst consistent set of axial and local peaking.

Analyses have shown that the operating limit MCPR's of 1.38 for 8 x 8 fuel and 1.36 for 7 x 7 fuel assure that the fuel cladding integrity safety limit is not exceeded during anticipated abnormal operational transients. Hence, we conclude that the above quoted operating limit MCPR's based on the limiting Rod Withdrawal Error Transient (further described in the section below) are acceptable.

3.3.3 Rod Withdrawal Error Transient

The rod withdrawal error transient is discussed in References 1 and 4 in terms of worst case conditions. Assumptions and descriptions of the rod withdrawal event are given in Reference 5. The information in these two references indicates that the local power range monitor subsystem (LPRM's) will detect high local powers and alarm. However, if the operator ignores the LPRM alarm, the rod block monitor subsystem (RBM) set at 105% of initial power level will terminate the transient while the critical power ratio is equal to 1.06 for 8 x 8 and 7 x 7 fuel. Therefore, as stated before, the rod withdrawal error which is the limiting transient for 8 x 8 and 7 x 7 fuel for the NMP-1 cycle 5 core still does not violate the safety limit MCPR and is acceptable. We conclude that the analyses and predicted consequences of this localized transient are acceptable as described in the previous sections.

3.3.4 Operating MCPR Limits for Less Than Rated Power and Flow

For the limiting transient of recirculation pump speed control failure at lower than rated power and flow conditions, the licensee will maintain the MCPR greater than the rated flow values (1.38 for 8 x 8 and 1.36 for 7 x 7 fuel) multiplied by the respective K_f factors appearing in Figure 3.1.7-1 of the Technical Specifications. The K_f factor curves were derived to assure that the most limiting transient occurring at less than rated flow will not exceed the safety limit MCPR of 1.06, and are therefore acceptable.

3.4 Accident Analysis

3.4.1 ECCS Appendix K Analaysis

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 and 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 cooling 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 results.

In Reference 8, the licensee submitted an evaluation of the NMP-1 ECCS performance for the Cycle 4 core. The licensee states that those same analyses are applicable to all fuel assemblies from the Cycle 4 core that will be part of the Cycle 5 core[9]. On the basis of the thermal hydraulic, nuclear, and mechanical similarity of the two cores, as already stated in previous sections, this is acceptable. Therefore, the staff conclusions on Cycle 4 (that the Cycle 4 core met all requirements of 10 CFR 50.46) which were reached after a detailed review of the break spectrum, single failures, break location, etc.,[10] also apply to the used fuel for the Cycle 5 core.

ECCS performance of the fresh 2.74% U-235 fuel to be loaded for the Cycle 5 core was analysed using the same acceptable methods[1, 9] as given in Reference 8. The results are reported in Reference 1. We therefore find the analyses acceptable for Cycle 5.

3.4.2 Steamline Break Accident

Steamline break accidents which are postulated to occur inside containment are covered by the ECCS analysis discussed in Section 3.4.1. The analysis of steamline break accidents occurring outside containment presented by the licensee by reference to NEDO-20360 is acceptable based on our generic review of NEDO-20360.[5]

3.4.3 Fuel Loading Error

Fuel loading errors are discussed in Reference 1, 3 and 4 for a fresh 2.74% U-235 low Gd content 8 x 8 fuel bundle placed in an improper location or rotated 180 degrees in a location near the center of the core. The information in those references indicates that the worst fuel loading is transposition of the fresh fuel intended for position (07,38) into position (25,28). This error results in a peak linear heat generation rate (LHGR) of 16.4 Kw/ft (1% plastic strain limit is approximately 22.5 Kw/ft) and a minimum critical power ratio (MCPR) equal to 1.11, which is above the SL-MCPR of 1.06 and is therefore acceptable. The licensee presented results of several other misloadings (References 3 and 4), thereby providing the bases for our conclusion that the misloading quoted above is the most severe.

3.4.4 Control Rod Drop Accident

The control rod drop accident for the NMP-1 Cycle 5 core is within the bounding analysis presented in Reference 5.[1] The Doppler coefficient of reactivity, the accident reactivity shape and magnitude function, and the rod drop scram reactivity functions are compared with the technical bases presented in Reference 5. This analysis is performed for Doppler coefficients of reactivity at the beginning of the Cycle 5 fuel cycle, at both cold (20°C) and hot (286°C) startup condition. It is shown by Figures 6-1, 6-2, 6-3, 6-4 and 6-5 of Reference 1 that the values of the parameters for this reloaded core are conservative with respect to the bounding values.

Therefore, we conclude that the consequences of a control rod drop accident (involving any in-sequence control rod during startup) will be below the design limit of 280 cal/gm.

3.4.5 Fuel Handling Accident

With respect to fuel handling accidents, in Reference 1 the applicant noted that the description and analyses of this event provided in the FSAR and discussed in the generic 8 x 8 reload report (Reference 5) are applicable to this reload. That is, the total activity released to the environment and the radiological exposures for the Cycle 5 core will be less than those values presented in the FSAR for the reference core. As identified in the FSAR the radiological exposures for this accident with the reference core are well below the guidelines set forth in 10 CFR 100. Therefore, it is concluded that the consequences of this accident for the Cycle 5 core will also be well below the 10 CFR 100 guidelines.

3.5 Overpressure Analysis

In Reference 1 the licensee presented the results of an overpressure analysis to demonstrate that an adequate margin exists below the ASME code allowable vessel pressure of 110% of vessel design pressure. The transient analyzed was the closure of all main steam isolation valves (MSIV) with no scram. The analysis was performed for 100% power with the end-of-cycle scram reactivity insertion rate curve, no scram, nuclear core physics parameters applicable to the end of Cycle 4 (EOC-4), and with no credit for the relief function of the safety/relief valves and with all safety valves operative. The results of this analysis for the worst time during cycle (i.e., worst void coefficient) indicate that the peak pressure at the vessel bottom would be 1334 psig. This provides a 41 psi margin to the vessel code limit of 1375 psig. (110% of the 1250 psig design pressure).

Overpressure analyses accepted by the NRC staff on other reload applications have assumed MSIV closure with high neutron flux scram and one failed safety valve. However, the assumption of no scram in the NMP-1 Cycle 5 overpressure analysis represents a conservatism which more than compensates for the assumption of no failed safety valve, and which compensates for any possible slight nonconservatism associated with small differences in nuclear core physics parameters between EOC-4 and the present Cycle 5 (which are not significantly different as stated in Section 3.1 above).

For the reasons stated above, the 41 psi pressure margin result of the overpressure protection analysis presented in Reference 1 for the Cycle 5 core is acceptable.

3.6 Thermal-Hydraulic Stability Analysis

A thermal-hydraulic stability analysis using the analytical methods discussed in Reference 5 was presented by the licensee for the NMP-1 Cycle 5 core.

The results show that the 7 x 7 and 8 x 8 channel hydrodynamic stability at the limiting (worst) condition (intersection of the natural circulation and extrapolated rod-block-curves) is within the channel operational design guide (decay ratio $\leq .5$). Calculations were also performed by the licensee to assess the reactor core power dynamic response at rated operating condition and at the low end of the flow control range (59% power and 40% flow). The results of these analyses show that the reactor core decay ratio is 0.030 at rated condition and 0.250 at the low end of the flow control range, which are both within the operational design guide of core decay ratio ≤ 0.25. These results are acceptable to the staff.

3.7 Peak Pressure Margin of 25 psi Below Lowest Set Safety Valve

The General Electric Company recommends that, during the worst overpressure abnormal operating transient, a 25 psi margin be maintained below the lowest safety valve setting. This is to prevent discharge of steam directly to the containment by the safety valves which can be avoided if the relief valves, whose discharges are piped to a position underwater in the torus, can handle the necessary steam flow. We find that the relief valves do meet this requirement.

The turbine trip without bypass transient analyses near end-of-cycle (EOC) for the Cycle 5 core show that the 25 psi margin can be maintained at full power until sometime after a burnup of 2000 Mwd/ton prior to EOC. At that burnup, a power "coast-down" will be initiated until 94% power is reached,[1, 2, 3] and power will then be maintained at a level not exceeding 94% until 1000 Mwd/ton burnup prior to end-of-cycle. An analysis was performed showing greater-than 25 psi pressure margin to the lowest safety valve setting exists at 94% power for exposures not exceeding 1000 Mwd/ton prior to EOC. At 1000 Mwd/ton prior to EOC, another power coast-down will be initiated until 92% power is reached and power will then be maintained at a level not exceeding 92% until EOC. An analysis at EOC was performed, showing greater than the 25 psi pressure margin to the lowest safety valve setting exists at 92% power for exposures to EOC.

We find the proposed power reductions described above[1, 2, 3] acceptable to preserve the 25 psi pressure margin to the lowest safety valve setting. The licensee has agreed to a License Restriction modification that requires the reactor to be operated in accordance with the power level profile given above and we have included these modifications in this amendment.

3.8 Change in Power Range for Use of Rod Worth Minimizer

The NMP-1 Cycle 5 core[1] references supplement 4 of NEDO-20360[5] for analysis of the rod drop accident. The referenced analysis takes credit for use of the rod worth minimizer (RWM) for powers below 20% rated thermal power to minimize the worst consequences of the rod drop accident. Nine Mile Point Unit No. 1 current Technical Specifications require use of the RWM to permit control motion only up to 10% of rated thermal power. Niagara Mohawk has proposed to change their Technical Specifications[11] to require use of the RWM for control rod motion up to 20% rated thermal power. This change is required to make the analysis and the operation consistent. This change improves plant safety and is therefore acceptable.

4.0 Technical Specification Changes

We find the proposed Technical Specification changes[2, 11] acceptable and consistent with the information in the Cycle 5 licensing submittal.[1] When the plant is operated in conformance to the proposed Technical Specification changes, it will be within the range of operating conditions assumed in the above described transients and accidents, which were found acceptable, and therefore such plant operation is acceptable.

On the basis of our review, described above, of the information provided on the Cycle 5 core, we conclude that the safety analyses are acceptable for the NMP-1 Cycle 5 core.

5.0 Reactor Water Level Limit During Maintenance

During the refueling outage, in preparation for Cycle 5 operation, feedwater nozzle inspection and maintenance work was performed in the reactor vessel. Since the facility does not currently have the capacity to fully unload the core, the inspection was performed with fuel in the reactor vessel. Technical Specifications in effect during the inspection and maintenance program permit the water level to be lowered to 293'9" elevation in order to clear the shielded platform. With the water level at that elevation, there is approximately 3'7" of water above the top of the core. Water is maintained in the core to remove the decay heat through the shutdown cooling system.

While the water level was being lowered, with no personnel in the vessel, a shutdown cooling pump tripped off causing a rapid decrease in water level from 294'9" to 293'6" elevation. The pump was immediately restarted and the water level was restored above 293'9" within seconds. The 293'6" elevation is the lowest elevation to which the water level could have drained because the shroud separates the core from the annular area, the recirculation pumps were isolated, and check valves in the shutdown cooling system prevent back flow of water from the core to the annulus.

All of the considerations presented in the safety evaluation permitting the water level to be lowered to 293'9" remain valid for the water level to be lowered to 293'6". No personnel exposures are involved, the temporary instrumentation systems are intact, the monitoring procedures are in force, and the core is more than adequately covered by cooling water.

We conclude that the lowering of the water level approximately 3 inches below the safety limit stated in the Technical Specifications during the refueling outage did not alter the reactor vessel or systems in any way, did not endanger personnel, was self limiting (i.e., the procedures would not have allowed the uncovering of the core), and that no danger to the public health and safety occurred during or following the event. Therefore, NMPC is authorized to resume operation of Nine Mile Point Unit No. 1.

6.0 Environmental Considerations

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.

Conclusions

We have concluded, based on the considerations discussed above, that: (1) because the change 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 change 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: June 27, 1977

REFERENCES

1) General Electric Boiling Water Reactor Reload No. 6 Licensing Amendment for Nine Mile Point Nuclear Power Station (Unit 1), NEDO-21466, Nov. 1976

2) Application for Technical Specification Changes submittal Dec. 3, 1976 for License No. DPR-63, Docket No. 50-220, with Attachments A and B.

3) Letter to Director of NRR from Gerald K. Rhode, Niagara Mohawk Power Corp., Responses to March 4, 1977 NRC Questions concerning NEDO-21466, March 14, 1977.

4) Verbal communications with S. Nowicki, NRC, from NMPC personnel, clarification of March 14 letter, March 28 and 29, 1977.

5) "General Electric Generic Reload Licensing Application for 8 x 8 Fuel," Revision 1, Including Supplements 1 through 4, April 1, 1976 (Supplement 4), NEDO-20360.

6) Status Report on the Licensing Topical Report "General Electric Boiling Water Generic Reload Application for 8 x 8 Fuel." NEDO-20360, Revision 1 and Supplement 1 by Division of Technical Review, Office of Nuclear Reactor Regulation, United States Nuclear Regulatory Commission, April 1975.

7) "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," NEDO-10958, 73 NED9, Class 1, November 1973.

8) Letter to Director of NRR from LeBoeuf, Lamb, Leiby and MacRae (Attorneys for Niagara Mowhawk Power Corp.), with attachments A and B, October 31, 1975.

9) Letter to Director of NRR, from James Bartlett, Niagara Mohawk Power Corp., March 31, 1977.

10) Safety Evaluation of the Nine Mile Point Unit 1 - Reload 5 by Reactor Safety Branch, V. Stello, Jr. (NRC) to K. Goller (NRC).

11) Application for Amendment to Technical Specifiction, Niagara Mohawk Corp., March 24,1977.

12) General Electric Boiling Water Reactor Reload-5 Licensing Submittal for Nine Mile Point Nuclear Power Station, Unit 1, NEDO-20772, August, 1975.

UNIT STATES NUCLEAR REGULATORY COM SION

DOCKET NO. 50-220

NIAGARA MOHAWK POWER CORPORATION

NOTICE OF ISSUANCE OF FACILITY LICENSE AMENDMENT

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 16 to Facility Operating License No. DPR-63 to the Niagara Mohawk Power Corporation (the licensee) which revised 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 consists of a change to the License Restriction and will modify the Technical Specifications to permit operation of the facility with 160 General Electric (GE) 8 x 8 reload fuel bundles and to require the use of the rod worth minimizer for power levels below 20% of rated thermal power.

The applications for the amendment comply 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 applications for amendment dated December 7, 1976 (supplemented by letter dated March 14, 1977) and March 24, 1977, (2) Amendment No. 16 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 City Library, 120 E. Second 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 27 day of June 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors