March 24, 1987

Docket No. 50-220

Mr. Charles V. Mangan
Senior Vice President
Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

Dear Mr. Mangan:

SUBJECT: LIMITING RELATIONSHIP BETWEEN CORE POWER AND CORE FLOW RATE (TAC 63532)

Re: Nine Mile Point Nuclear Station, Unit No. 1

The Commission has issued the enclosed Amendment No. 92 to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station, Unit No. 1. This amendment is in response to your application dated October 30, 1986, as supplemented January 15, 1987.

The amendment modifies Figures 2.1.1 and 3.1.7aa of the Appendix A Technical Specifications (TS) regarding the limiting relationships between core power and core flow rate. Specifically, the changes to Figures 2.1.1 and 3.1.7aa reflect changes to: (1) reduce a restriction on the reactor operating range and (2) provide consistency with the current reload analyses for the present Nine Mile Point Unit 1 core configuration and parameters.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notices.

Sincerely,

Original signed by

Rajender Auluck, Acting Director
BWR Project Directorate #1
Division of BWR Licensing

Enclosures:
1. Amendment No. 92 to License No. DPR-63
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. C. V. Mangan
Niagara Mohawk Power Corporation

Nine Mile Point Nuclear Station,
Unit No. 1

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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. 92
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 October 30, 1986, as supplemented January 15, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 92 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Rajender Auluck

Rajender Auluck, Acting Director
BWR Project Directorate #1
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 24, 1987

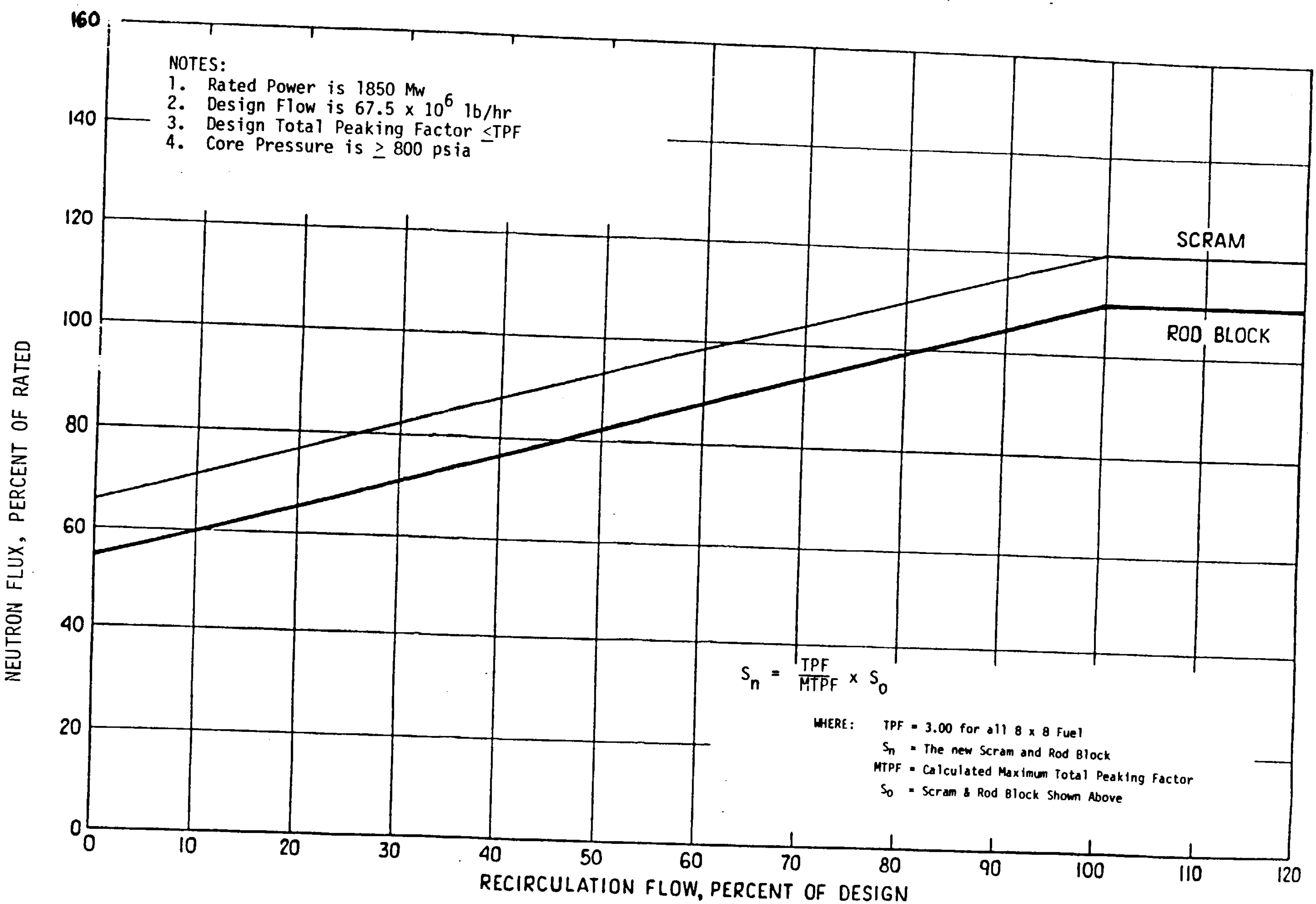
ATTACHMENT TO LICENSE AMENDMENT NO. 92

FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
8	8
18	18
64e	64e
70a	70a
70d	70d



$$S_n = \frac{TPF}{MTPF} \times S_o$$

WHERE: TPF = 3.00 for all 8 x 8 Fuel
S_n = The new Scram and Rod Block
MTPF = Calculated Maximum Total Peaking Factor
S_o = Scram & Rod Block Shown Above

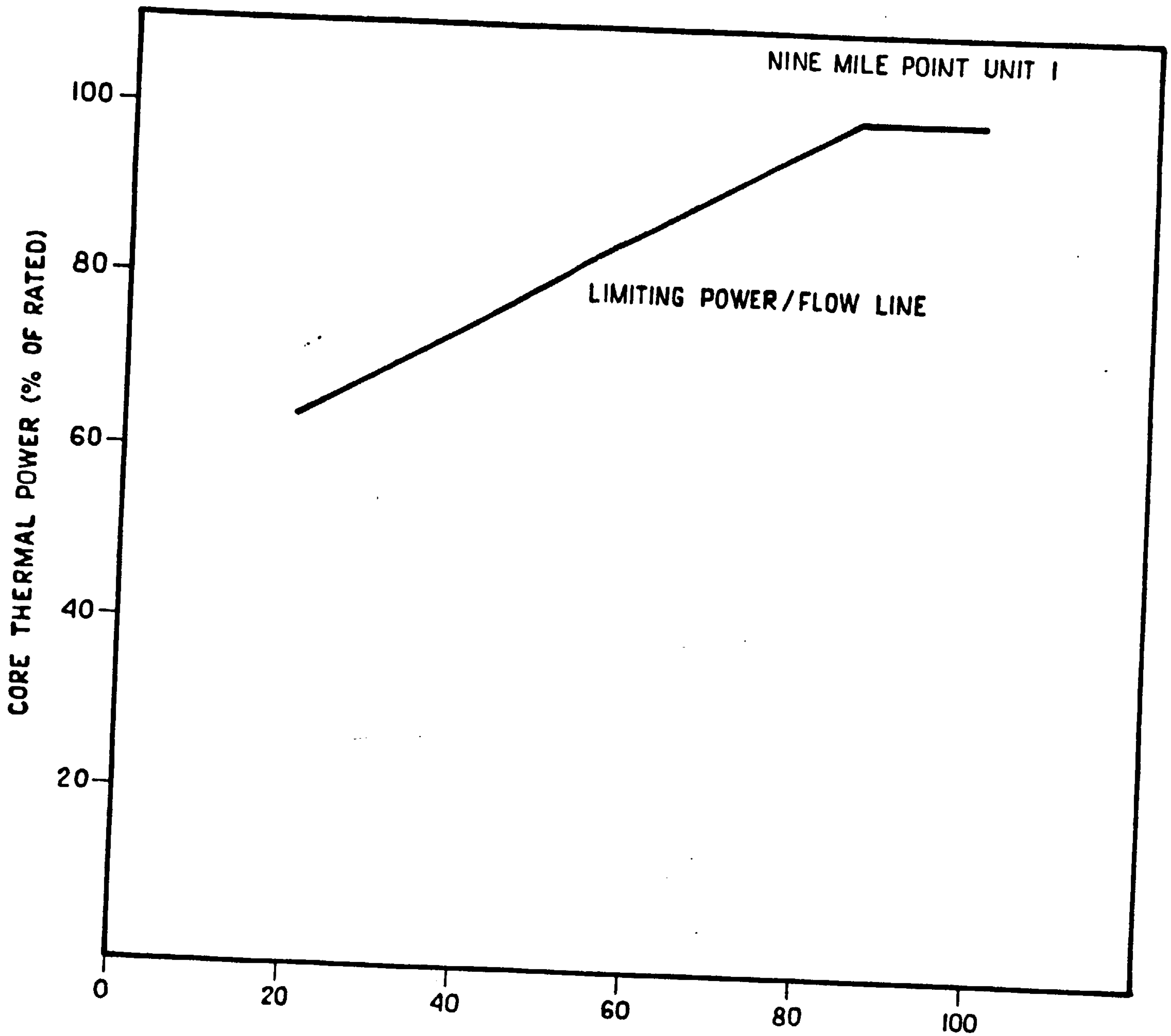
FIGURE 2.1.1 FLOW BIASED SCRAM AND APRM ROD BLOCK steady-state operation is at 110% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum total peaking factor exceeds the design peaking factor, thus, preserving the APRM rod block safety margin.

g-h. The low pressure isolation of the main steam lines at 850 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line isolation on reactor low pressure and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at $\leq$10% valve closure, there is no increase in neutron flux and peak pressure in the vessel dome is limited to 1141 psig. (8, 9, 10).

The operator will set the pressure trip at greater than or equal to 850 psig and the isolation valve stem position scram setting at less than or equal to 10% of valve stem position from full open. However, the actual pressure set point can be as much as 15.8 psi lower than the indicated 850 psig and the valve position set point can be as much as 2.5% of stem position greater. These allowable deviations are due to instrument error, operator setting error and drift with time.

In addition to the above mentioned LS^3, other reactor protection system devices (LCO 3.6.2) serve as a secondary backup to the LS^3 chosen. These are as follows:

High fission product activity released from the core is sensed in the main steam lines by the high radiation main steam line monitors. These monitors provide a backup scram signal and also close the main steam line isolation valves.



Amendment No. 23, 39, 41, 92

Figure 3.1.7.aa LIMITING POWER FLOW LINE

of the plant, a MCPR evaluation will be made at the 25% thermal power level with minimum recirculation pump
speed. The MCPR margin will thus be demonstrated such that future MCPR evaluations below this power level will
be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is
sufficient since power distribution shifts are very slow when there have not been significant power or control
rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that
MCPR will be known following a change in power or power shape (regardless of magnitude) that could place
operation at a thermal limit.

Figure 3.1.7-1 is used for calculating MCPR during operation at other rated conditions. For the case of
automatic flow control, the K_f factor is determined such that any automatic increase in power (due to flow
control) will always result in arriving at the nominal required MCPR at 100% power. For manual flow control, the
K_f is determined such that an inadvertent increase in core flow (i.e, operator error or recirculation pump
speed controller failure) would result in arriving at the 99.9% limit MCPR when core flow reaches the maximum
possible core flow corresponding to a particular setting of the recirculation pump MG set scoop tube maximum
speed control limiting set screws. These screws are to be calibrated and set to a particular value and whenever
the plant is operating in manual flow control the K_f defined by that setting of the screws is to be used in the
determination of required MCPR. This will assure that the reduction in MCPR associated with an inadvertent flow
increase always satisfies the 99.9% requirement. Irrespective of the scoop tube setting, the required MCPR is
never allowed to be less than the nominal MCPR (i.e. K_f is never less than unity).

Power/Flow Relationship

The power/flow curve is the locus of critical power as a function of flow from which the occurrence of abnormal
operating transients will yield results within defined plant safety limits. Each transient and postulated
accident applicable to operation of the plant was analyzed along the power/flow line. The analysis (7, 8, 12,
14) justifies the operating envelope bounded by the power/flow curve as long as other operating limits are
satisfied. Operation under the power/flow line is designed to enable the direct ascension to full power within
the design basis for the plant.

REFERENCES FOR BASES 3.1.7 AND 4.1.7 FUEL RODS

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

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

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

(4) "General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel," NEDO-20360, Supplement 1 to Revision 1, December 1974.

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

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

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

(8) Licensing Topical Report General Electric Boiling Water Reactor Generic Reload Fuel Application, NEDE-24011-P-A, August 1978.

(9) Final Safety Analysis Report, Nine Mile Point Nuclear Station, Niagara Mohawk Power Corporation, June 1967.

(10) NRC Safety Evaluation, Amendment No. 24 to DPR-63 contained in a lettter from George Lear, NRC, to D. P. Dise dated May 15, 1978.

(11) "Core Flow Distribution in a General Electric Boiling Water Reactor as Measured in Quad Cities Unit 1," NEDO-10722A.

(12) Nine Mile Point Nuclear Power Station Unit 1, Extended Load line Limit Analysis, License Amendment Submittal (Cycle 6), NEDO-24185, April 1979.

(13) Loss of Coolant Accident Analysis Report for Nine Mile Point Unit One Nuclear Power Station, NEDO-24348, August 1981.

(14) General Electric Boiling Water Reactor Extended Load Line Limit Analysis for Nine Mile Point 1 Cycle 9, NEDC-31126, February 1986.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 92 TO FACILITY OPERATING LICENSE NO. DPR-63

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-220

1.0 INTRODUCTION

By application dated October 30, 1986, as supplemented January 15, 1987, Niagara Mohawk Power Corporation (the licensee) requested an amendment to Appendix A of Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station, Unit No. 1 (NMP-1). The amendment would modify Figures 2.1.1 and 3.1.7.aa in the Appendix A Technical Specifications (TS) regarding the limiting relationships between core power and core flow rate. Specifically, the changes to Figures 2.1.1 and 3.1.7aa would reflect changes to: (1) reduce a restriction on the reactor operating range, and (2) provide consistency with the current reload analyses for the present Nine Mile Point Unit 1 core configuration and parameters. In support of the proposed modification, the licensee provided General Electric Company (GE) Report, NEDC-31126, "General Electric Boiling Water Reactor Extended Load Line Limit Analysis for Nine Mile Point 1 Cycle 9" dated February 1986.

2.0 EVALUATION

Presently, the NMP-1 TS limit full power operation to the flow range from 91% to 100%. The Extended Load Line Limit Analysis (ELLLA) provides the basis for reactor operation in the region bounded by the 108% average power range monitor (APRM) rod block line (See Figure 2.1.1) and the licensed power. The present Power/Flow line was reviewed and approved in the Staff's Safety Evaluation supporting Amendment No. 36 to Facility Operating License No. DPR-63 dated March 29, 1980. The requested change affects a relatively small triangular area of the Power/Flow Map for core power between 85 and 100% and core flow between 61 and 91%. A GE safety analysis (NEDC-31126, February 1986) was performed to demonstrate that the consequences of transients and accidents initiated from within the expanded ELLLA region are bounded by the consequences of the same events initiated from the licensing basis condition for NMP-1, Cycle 9. Consideration was also given to overpressure protection for compliance with the American Society of Mechanical Engineers (ASME) Pressure Vessel Code, and reactor core thermal-hydraulic stability.

Principal considerations in the review of the expanded ELLLA for NMP-1 include:

1. The slope of the rod block line has been previously reviewed and approved for NMP-1 (Amendment No. 36, March 28, 1980) and is generally applicable.

2. The limiting transients initiated within the proposed expanded ELLLA region are still limiting at 100% power/100% flow (See Table 3-2 of GE NEDC-31126, February 1986). Therefore, no change in the present operating limit minimum critical power ratio (MCPR) of 1.40 is required.

3. The effect of less than rated flow on loss-of-coolant accident analyses has been generically reviewed and approved for referencing (NRC Letter to GE dated May 19, 1978). No changes to the present maximum average planar linear heat generation rate (MAPLHGR) limits are required.

4. The calculated peak pressure for main steam isolation valve closure without scram increases by 1 psi which results in a value 46 psi below the allowable 1375 psi ASME Pressure Vessel Code limit for overpressure protection and thus remains acceptable.

5. The proposed extended ELLLA region is beyond the region of thermal-hydraulic stability (THS) concerns; therefore the THS requirements identified in NRC Generic Letter 86-02 (January 23, 1986) remain satisfied.

Based on our review of the anticipated changes in core behavior, we conclude that the proposed changes correctly address the proposed extended operating region and are, therefore, acceptable. The revised figures reflect the new flow-biased scram and APRM rod block settings and the limiting power flow line. No changes to the present TS operating limit Minimum Critical Power Ratio or MAPLHGR limits are required.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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 nor to the health and safety of the public.

Principal Contributor: M. McCoy

Dated: March 24, 1987

March 24, 1987

MEMORANDUM FOR: Sholly Coordinator

FROM: Rajender Auluck , Acting Director
BWR Project Directorate #1, DBL

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE (TAC 63532)

Niagara Mohawk Power Corporation, Docket No. 50-220, Nine Mile Point Nuclear Station, Unit No. 1, Oswego County, New York

Date of amendment request: October 30, 1986 as supplemented by letter dated January 15, 1987.

Brief description of amendment: The amendment modifies Figures 2.1.1 and 3.1.7aa of the Appendix A Technical Specifications regarding the limiting relationships between core power and core flow rate.

Date of issuance: March 24, 1987

Effective date: March 24, 1987

Amendment No.: 92

Facility Operating License No. DPR-63. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: January 28, 1987 (52 FR 2884).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 24, 1987 .

No significant hazards consideration comments received: No.

Local Public Document Room location: State University of New York, Penfield Library, Reference and Documents Department, Oswego, New York 13126.

Original signed by
Rajender Auluck , Acting Director
BWR Project Directorate #1, DBL

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