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. 36
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 June 28, 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, 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. 36 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Beyond the point in the Cycle 6 fuel cycle at which the reactivity reduction rate during a scram is less than that of the curve marked EOC 6 minus 1500 Mwd/T in Figure 2C of "Supplemental Reload Licensing Submittal for Nine Mile Point Nuclear Power Station Unit 1 Reload No. 1, Ring Reanalysis Supplement," NEDO 24155-1 Supplement 1 dated December 1978, operation of the reactor shall not exceed a core thermal power of 1813 megawatts (98% of rated) at rated flow conditions.

Beyond the point in the Cycle 6 fuel cycle at which the reactivity reduction rate during a scram is less than that of the curve marked EOC 6 minus 1000 Mwd/T in Figure 2B of "Supplemental Reload Licensing Submittal for Nine Mile Point Nuclear Power Station (Unit 1) Reload No. 7," NEDO 24155, 78NED291, dated November 1978, operation of the reactor shall not exceed a core thermal power of 1757 megawatts (95% of rated) at rated flow conditions.

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

T. A. Ippolito

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 28, 1980

9t

ATTACHMENT TO LICENSE AMENDMENT NO. 36

FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

Revise Appendix A by removing the following pages and replacing with the attached identically numbered pages. Marginal lines indicate area of change.

15
20
64c
70a
70c

chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses (5,6,8,9,10,11,13) demonstrate that with a 120% scram trip setting, none of the abnormal operational transients analyzed violate the fuel safety limit and there is a substantial margin from fuel damage.

However, in response to expressed beliefs (7) that variation of APRM flux scram with recirculation flow is a prudent measure to assure safe plant operation during the design confirmation phase of plant operation, the scram setting will be varied with recirculation flow.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity safety limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity safety limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MTPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Figure 2.1.1 when the maximum total peaking factor is greater than the limiting total peaking factor.

b. Normal operation of the automatic recirculation pump control will be in excess of 30% rated flow; therefore, little operation below 30% flow is anticipated. For operation in the startup mode while the reactor is at low pressure, the IRM scram setting is 12% of rated neutron flux. Although the operator will set the IRM scram trip at 12% of rated neutron flux or less, the actual scram setting can be as much as 2.5% of rated neutron flux greater. This includes the margins discussed above. This provides adequate margin between the setpoint and the safety limit at 25% of rated power. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. There are a few possible sources of rapid reactivity input to the system in the low power flow condition. Effects of increasing pressure at zero or low

(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.

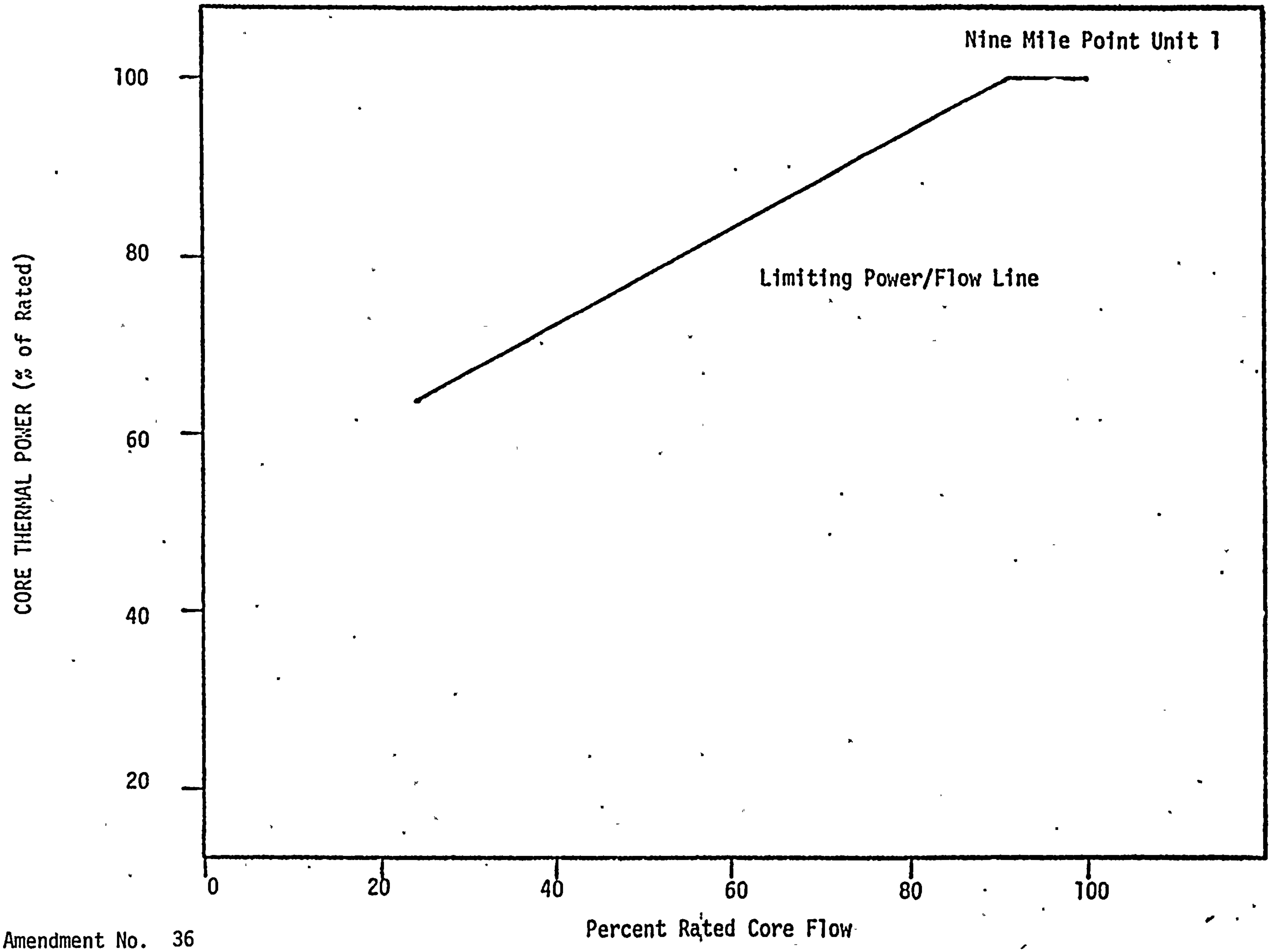
(9) Letter, T. J. Brosnan, Niagara Mohawk Power Corporation, to Peter A. Morris, Division of Reactor Licensing, USAEC, dated February 28, 1972.

(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.

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



Amendment No. 36

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

Power/Flow Relationship

The power/flow curve is the locus of critical power as a function of flow from which the occurrence of abnormal operating transients will yield results within defined plant safety limits. Each transient and postulated accident applicable to operation of the plant was analyzed along the power/flow line. The analysis (7,8,9) 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.

Reactor power level in the one-loop-isolated mode is restricted to a power level which has been analyzed and found acceptable.

REFERENCES FOR BASES 3.1.7 AND 4.1.7 FUEL RODS

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

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

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

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

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

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

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

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

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