4/7/78

Distribution
Docket
ORB #3
Local PDR
NRC PDR
VStello
GLEar
RClark
SSheppard
Attorney, OELD
OI&E (5)
BJones (4)
BScharf (10)
JMcGough
DEisenhut
ACRS (16)
DRoss
TBAbernathy
JRBuchanan

Docket No. 50-220

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.   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 is in response to your request dated July 18, 1977. In reviewing your application it was found that certain changes in your proposed Technical Specifications were required. These changes were discussed with and approved by your staff.

The amendment modifies the Technical Specifications by (1) modifying the Flow Biased Scram and APRM Rod Block curve; (2) replacing assembly averaged power-void relationships with a limiting power/flow curve; and (3) increasing the operating Minimum Critical Power Ratio (MCPR) for 7x7 fuel. The amendment also deletes the current Specification 6.12, Respiratory Protection Program in accordance with our letter dated July 29, 1977.

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

Sincerely,

Original signed by

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

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

cc w/enclosures:
See page 2

Subject to changes noted on "working copy"

OFFICE	ORB #3	ORB #3	OELD	ORB #3		
SURNAME	SSheppard	RClark:mjf	BnBordenick	GLear		
DATE	3/21/78	3/22/78	3/24/78	3/ /78		

NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 – 626-624

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

April 7, 1978

Docket No. 50-220

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. 23 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 is in response to your request dated July 18, 1977. In reviewing your application it was found that certain changes in your proposed Technical Specifications were required. These changes were discussed with and approved by your staff.

The amendment modifies the Technical Specifications by (1) modifying the Flow Biased Scram and APRM Rod Block curve; (2) replacing assembly averaged power-void relationships with a limiting power/flow curve; and (3) increasing the operating Minimum Critical Power Ratio (MCPR) for 7x7 fuel. The amendment also deletes the current Specification 6.12, Respiratory Protection Program in accordance with our letter dated July 29, 1977.

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

Sincerely,

George Lear

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

Enclosures:
1. Amendment No. 23 to DPR-63
2. Safety Evaluation
3. Notice

cc w/enclosures:
See page 2

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Chief, Energy Systems Analysis Branch (AW-459)
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New York, New York 10007

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46 E. Bridge Street
Oswego, New York 13126

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. 23
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 July 18, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 23, 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

George Lear

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 7, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 23

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 contain vertical lines indicating the area of change. Delete pages 64d thru 64f and pages 264 thru 267.

Remove	Replace
8	8
20	20
64a	64a
64b	64b
64c	64c
64d thru 64f	---
70a thru 70c	70a thru 70c
260 thru 263	260 thru 263
264 thru 267	---
268	264

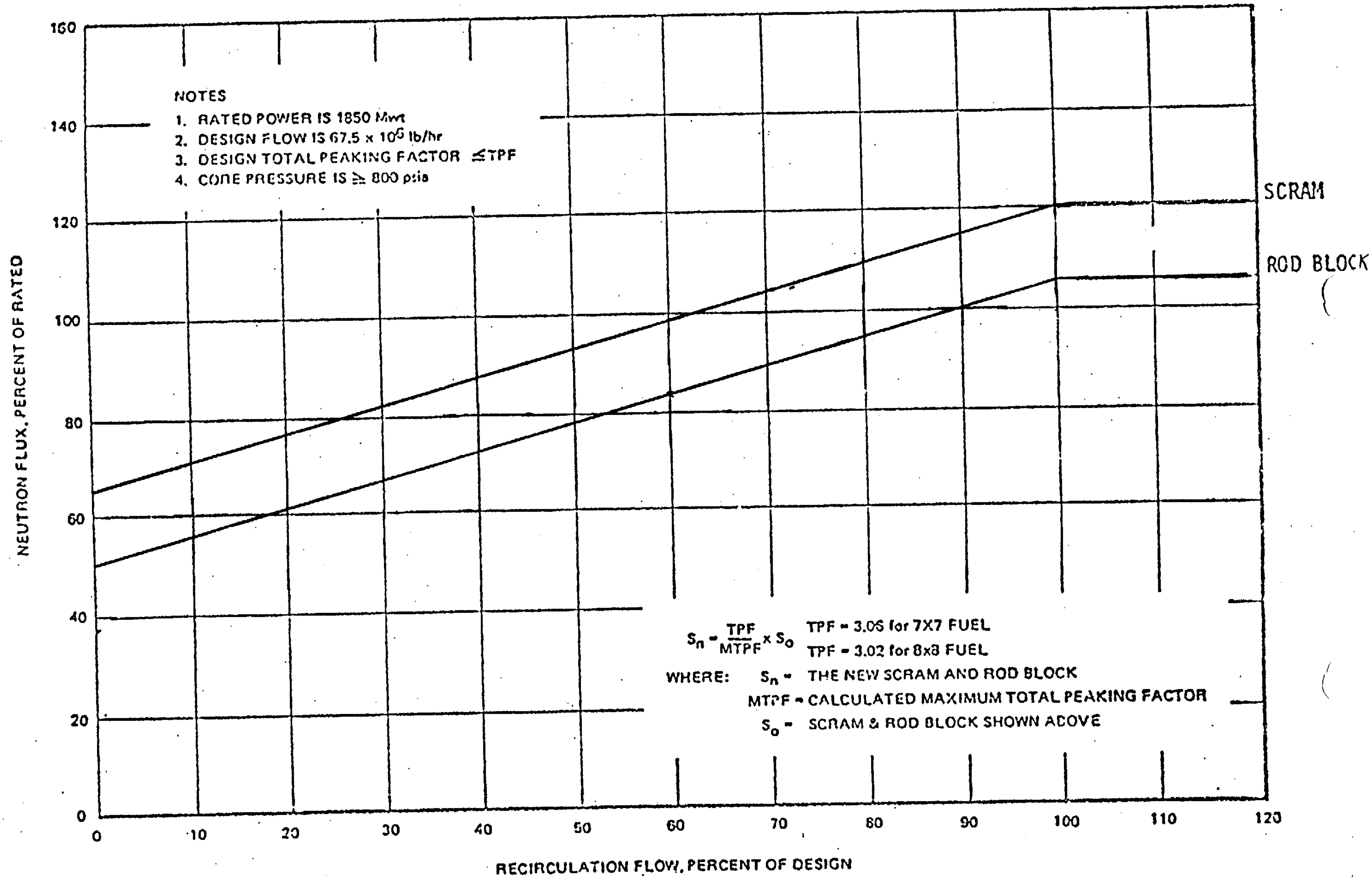


Figure 2.1.1. Flow Biased Scram and APRM Rod Block

(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.

LIMITING CONDITIONS FOR OPERATION

c. Minimum Critical Power Ratio (MCPR)

During power operation MCPR shall be $\geq$ 1.37 for 7x7 fuel and $\geq$ 1.38 for 8x8 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 the operating MCPRs are 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.

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.

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 CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

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, 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.

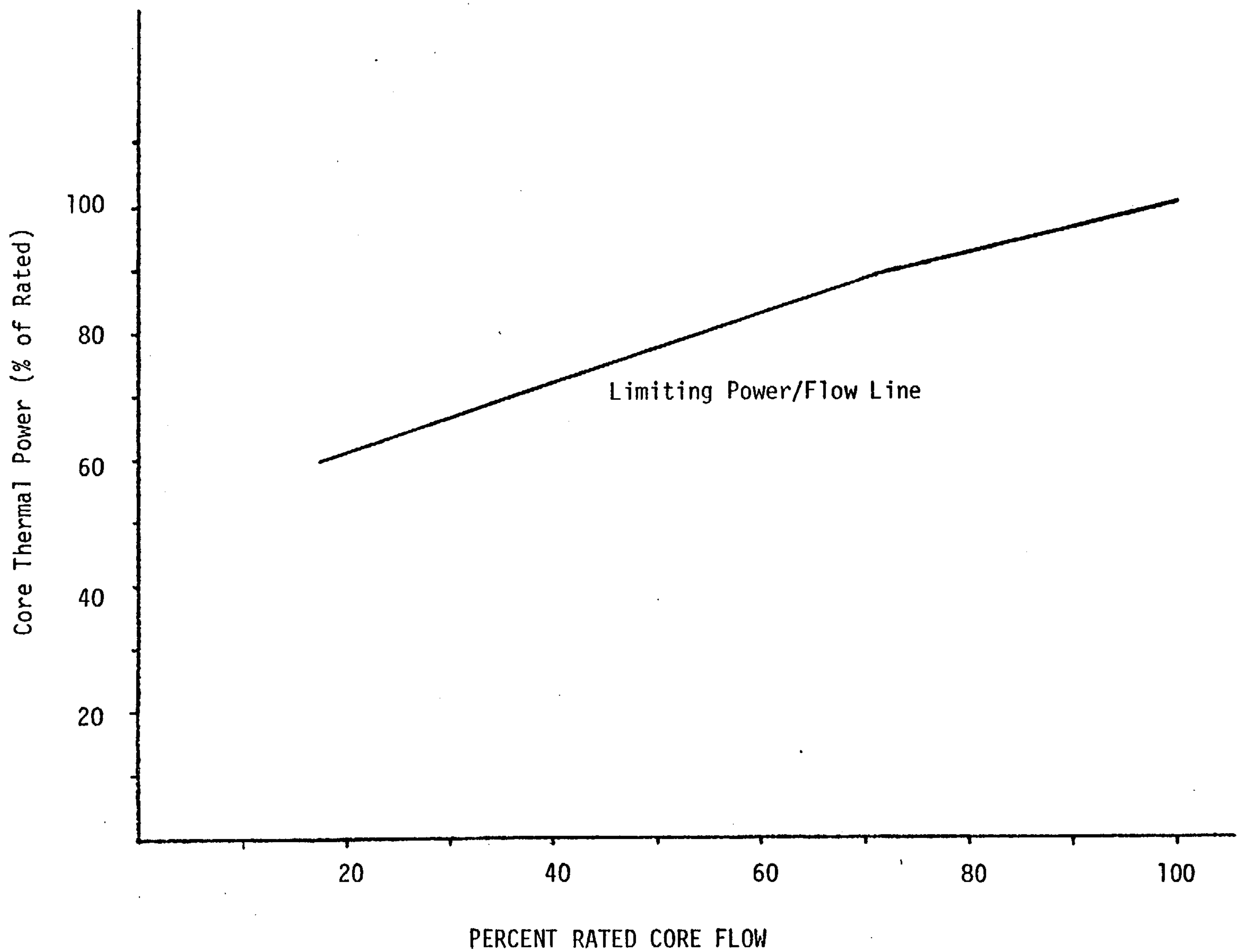
e. Reporting Requirements

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 3.1.7.aa

NINE MILE POINT UNIT 1

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] 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.

(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.

Reporting Requirements

The LCO's associated with monitoring the fuel rod operating conditions are required to be met at all times, i.e., there is no allowable time in which the plant can knowingly exceed the limiting values of MAPLHGR, LHGR, MCPR, or Power/Flow Ratio. It is a requirement, as stated in Specifications 3.1.7.a, b, c & d that if at any time during power operation, it is determined that the limiting values for MAPLHGR, LGHR, MCPR, or Power/Flow Ratio are exceeded, action is then initiated to restore operation to within the prescribed limits. This action is initiated as soon as normal surveillance indicates that an operating limit has been reached. Each event involving operation beyond a specified limit shall be reported as a Reportable Occurrence. If the specified corrective action described in the LCO's was taken, a thirty-day written report is acceptable.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the SORC and the SRAB.

6.11 Radiation Protection Program

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

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6.13 High Radiation Area

6.13.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20:

a. Each High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.13.1(a) above, and in addition locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty.

6.14 Fire Protection Inspection

6.14.1 An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified off-site licensee personnel or an outside fire protection firm.

6.14.2 An inspection and audit by an outside qualified fire consultant shall be performed at intervals no greater than 3 years.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 23 TO FACILITY OPERATING LICENSE NO. DPR-63

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT UNIT NO. 1

DOCKET NO. 50-220

1.0 Introduction

1.1 By letter dated July 18, 1977, Niagara Mohawk Power Corporation (NMPC) requested an amendment to Facility Operating License No. DPR-63 for the Nine Mile Point Unit No. 1 (NMP-1). The amendment would modify the Technical Specifications by: (1) modifying the Flow Biased Scram and Average Power Range Monitor (APRM) Rod Block Curve; (2) replacing assembly averaged power-void relationship curves with a limiting power/flow curve; and (3) increasing the operating Minimum Critical Power Ratio (MCPR) for 7x7 fuel.

1.2 In addition to the requested modification to the Technical Specifications this amendment makes administrative changes to the Respiratory Protection Program Specification 6.12. We advised NMPC, by our letter dated July 29, 1977, that pursuant to 10 CFR 20.103(c) and (f), if they desire to receive credit for use of respiratory protective equipment at their facility after December 27, 1977, such use must be as stipulated in Regulatory Guide 8.15 rather than as specified in their current Technical Specifications. Based on the revocation provision of the NMP-1 current specification on respiratory protection we are deleting Specification 6.12. NMPC was advised that no response to our letter was required.

2.0 Discussion

2.1 NMPC proposes to change the Flow Biased Scram and APRM Rod Block curve for the NMP-1 to provide operational flexibility. In addition to modifying the Rod Block and Scram curve, NMPC proposes to use a power/flow curve as the locus of limiting allowable thermal power as a function of flow. By using these power and flow conditions as constraints to power operation the power-void relationships are satisfied and the current Assembly Averaged Power-Void Relationships using B factors (Power-Void Limits) can be removed from the Technical Specifications. All reactor safety analyses are based on these power and flow constraints and the power/flow-condition are such that the occurrence of an abnormal operating transient will result in operation still within safety limits. The basis for these proposed changes are provided in Reference 1.

Minor administrative changes were made to page 70b, Bases for 3.1.7 and 4.1.7 Fuel Rods, and were agreed to by NMPC. The changes were made to make it clear in the Bases that limiting conditions for operation are specified for 4 parameters: MAPLHGR, LHGR, MCPR and Power/Flow Ratio.

2.2 Our letter of July 29, 1977 to NMPC was a generic letter sent to all licensees with a revocation provision of their specification on respiratory protection. The deletion of Specification 6.12 is being handled as an administrative change.

3.0 Evaluation

The modifications in the APRM rod block and the flux scram lines provide a power/flow envelope which satisfies the same minimum safety criteria as previously reviewed by the NRC staff and found acceptable in the Safety Evaluation Report associated with license amendment No. 16, dated June 17, 1977. NMPC has provided the results of analyses and sensitivity studies to demonstrate that these criteria were met. The consequences of any abnormal transient or accident has been verified to be within safety limits previously found acceptable in the Safety Evaluation Report dated June 17, 1977, noted above.

3.1 Transients

As shown in Reference 2, the two most limiting abnormal operational transients for the NMP-1 are Turbine Trip With-Failure-of-the-Bypass-Valves (TTWOB) and Rod Withdrawal Error (RWE). The transient analyses (Reference 1) and sensitivity studies for the proposed changes were performed with the same input parameters as those for the Reload 6 analyses (Reference 2) except for void coefficient. This new analysis used a "design conservatism factor" of 1.33 on void coefficient as compared to the 1.25 used in the reload analyses. The use of the 1.33 design factor results in a more conservative analysis. Because the end-of-cycle 5 (EOC5) scram reactivity insertion function is the most limiting condition, this curve was used for all analyses. Each transient was analyzed at power/flow conditions of 100%/100%, 91%/75%, and 85%/61% to provide verification of transient behavior along the rod intercept line to the point of rated power and flow. At the rated power/flow point the resultant transient behavior is the same as the previous analysis because the trip and rod block functions were not changed. The ΔCPR (change in Critical Power Ratio) derived at the two lower values of power/flow are less than the ΔCPR for rated conditions for all transients except RWE. Since the RWE at the 91%/75% point resulted in a slightly higher ΔCPR (0+.01) than at the 100%/100% point for 7x7 fuel, NMPC proposed to increase the 7x7 operating Minimum Critical Power Ration (MCPR) limit by 0.01. For Cycle 5, the operating MCPR for 7x7 fuel is changed from $\geq$1.36 to $\geq$1.37. Since this increase in MCPR for 7x7 fuel compensates for the higher ΔCPR resulting from the RWE analysis, we find this change necessary and acceptable.

3.1.1 Rod Withdrawal Error

Since it is not apparent the RWE will not be the more limiting at lower power levels along the rod block intercept line, the RWE was analyzed along the rod block intercept line at the rod block intercept point (the 85%/61% point), the 100%/100% point and an intermediate point for the current cycle. A similar analyses must be performed or appropriate justification provided for future cycles.

At the 85%/61% point the RWE results in a ΔCPR that is 0.02 higher in value than at the 100%/100% point. The K_f factor which is multiplied times MCPR is normally used to provide margin for flow increase transients and will be at least 1.07 at the 85%/61% point. The product of 1.07 and the MCPR operating limit at the 85%/61% point is high enough to more than compensate for the 0.02 increase in ΔCPR. This compensation has been previously found acceptable (Reference 6) and is applicable for NMP-1.

The APRM rod block setpoint is selected to allow for failed instruments for the worst allowable power profile. It is demonstrated that even if the operator ignores all alarms during the course of this transient, the rod blocks will stop rod withdrawal when the CPR is 1.06 (the CPR safety limit). At powers and flows lower than the 85%/61% condition within the proposed operating envelope, a RWE results in smaller ΔCPR values. The use of the present K_f factors limit the control rod position such that the resulting MCPR's are conservative and bound the ΔCPR due to a RWE. The consequences of the RWE transient decrease at lower flows and the effective MCPR required by the use of the K_f values become increasingly conservative.

3.1.2 Peak Pressure Margin (25 psi Below Lowest Set Safety Value)

An analysis of the transient which involves main steam line isolation valve (MSIV) closure with high flow scram is used to evaluate compliance with the ASME pressure vessel code. The GE design requirement for adequacy of the safety valve capacity is a 25 psi margin between the peak vessel pressure and the ASME Boiler and Pressure Vessel code limit of 1375 psig based on a postulated MSIV closure transient without a scram. As the result of the postulated transient, the peak vessel bottom pressure at the rod block intercept point (85%/61%) is 1315 psig, 15 psi below that for the 100%/100% analysis and 60 psi below the code limit; this result is acceptable.

3.1.3 Operating MCPR Limits for Less Than Rated Power and Flow

A statistical analysis was performed to determine the part-load safety limit MCPR requirements along the APRM rod block line (Reference 4). The results of the analysis show a small increase (.01 at the rod block intercept point) in the safety limit MCPR requirement for part-load conditions due to increase in uncertainty of flow measurement.

However, this small increase in the part-load safety limit is more than compensated for by the K_f factor-based operating MCPR limit for part-load conditions. (For NMP-1, the operating MCPR increases by about 0.09 due to the K_f factor, compared to a 0.01 decrease due to flow measurement uncertainty). The K_f factor is determined such that any inadvertent increase in core flow results in a MCPR greater than or equal to the safety limit MCPR at 100% power. Therefore, the safety limit will be satisfied when the plant is operating in accordance with the new power-flow operating envelope and is acceptable.

3.1.4 Xenon Transients

The safety analyses normally performed assume equilibrium xenon conditions. To investigate the change in MCPR from non-equilibrium conditions, a localized xenon transient was simulated. On the basis of actual rod swap results, a reactivity swing from equilibrium to peak of $-0.05 \Delta k/k$ was conservatively established to model the effect of xenon. This reactivity swing was used as input as a local reactivity change in the BWR simulator and the associated change in MCPR was calculated (Reference 3). The analysis was performed assuming control rods in and control rods out of the reactor core. This conservative, xenon induced reactivity change assuming the extremes of rod configurations has given a calculated Δ MCPR of -0.09 in the limiting bundle. This change results in a change of operating MCPR from 1.36 to 1.27 in the worst case. The calculations provided in Reference 3 show that safety limits are not approached during xenon transients, and since this analysis provides conservative estimate of xenon transient effects, we find these results acceptable.

3.2 Thermal-Hydraulic Stability Analysis

Thermal-hydraulic stability analyses for NMP-1 Cycle 5 are presented in Reference 2. These results cover both the current 105% rod block and the modified extrapolated rod block line at 108%. The Reference 2 results are bounded by this analysis and are acceptable.

3.3 Accident Analysis

Currently, power void limits (B-Factors) are utilized in the NMP-1 Technical Specifications to assure compliance with the assembly void fraction assumption used in the Loss of Coolant Accident (LOCA) analysis. These power void limits in the Technical Specifications are no longer necessary when the proposed power/flow limit line is added to the Technical Specifications since the combination of MCPR operating limits and the proposed power/flow limit line provide adequate assurance that operating conditions will be more conservative than the initial conditions assumed in the LOCA analysis for NMP-1.

An evaluation of the void fraction, post LOCA was performed to determine effects of voiding on rod dry-out during the accident. For the LOCA analysis, the initial MCPR value used was 1.19, and the bundle axial power shape was varied in both position and peaking factor in order to determine a conservative void fraction consistent with this MCPR. The Technical Specifications on MCPR operating limits at rated conditions are in excess of 1.3; hence, these higher MCPR values would result in void fraction lower than those calculated from the MCPR of 1.19 used in the analysis. Since the operating MCPR at reduced power and flow must be increased in accordance with the K_f factor in the Technical Specifications this also contributes to a lower void fraction than used in the analysis. The time to "rod dryout" is directly proportional to the volume of water initially in the fuel bundle; thus, a lower void fraction would result in a longer time to dryout the bundle and, therefore, the current analysis basis is conservative.

The LOCA analysis performed at the 100% power/100% flow point has been found to be limiting in relation to other points along the proposed power/flow curve. The operation of NMP-1 within the envelope bounded by the proposed power/flow curve assures that the current LOCA analysis basis is conservative with respect to all operating conditions and therefore is acceptable.

3.4 Summary

NMPC has shown the changes in APRM rod block and flux scram curves do not significantly affect the consequences for any transient or accident previously analyzed and accepted by the NRC. On this basis the proposed Technical Specification change is found acceptable.

4.0 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.

5.0 Conclusions

We have concluded, based on the considerations discussed above, 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 7, 1978

References

1. "Nine Mile Point Nuclear Power Station Unit 1 Load Line Limit Analysis License Amendment Submittal," NEDO 24012, May 1977.

2. "General Electric Boiling Water Reactor Reload 6 Licensing Amendments for Nine Mile Point Nuclear Power Station Unit 1," NEDO 21466.

3. "Millstone Point Nuclear Power Station Unit 1 Load Line Limit Analysis License Amendment Submittal," NEDO 21285, June 1976.

4. Final Safety Analysis Report, Millstone Nuclear Power Station Unit 1, Docket No. 50-245, License No. DPR-21.

5. "General Electric Thermal Analysis Basis," NEDO 10958, January 1977.

6. Memo to K. Goller from V. Stello, "Review of Millstone Unit 1 Reload 3 TAR 1775," dated October 30, 1975.

UNITED STATES NUCLEAR REGULATORY COMMISSION

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. 23 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 modifies the Technical Specifications by (1) modifying the Flow Biased Scram and APRM Rod Block curve; (2) replacing assembly averaged power-void relationships with a limiting power/flow curve; and (3) increasing the operating Minimum Critical Power Ratio (MCPR) for 7x7 fuel. The amendment also deletes the current Specification 6.12, Respiratory Protection Program in accordance with the Commission's letter to the licensee dated July 29, 1977.

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, negative declaration or 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 July 18, 1977, (2) the Commissions related letter dated July 29, 1977, (3) Amendment No. 23 to License No. DPR-63, and (4) 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, 46 E Bridge Street, Oswego, New York 13126. A copy of items (2), (3) and (4) 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 7th day of April 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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