JUL 16 1974

Docket No. 50-220

Niagara Mohawk Power Corporation
ATTN: Mr. R. R. Schneider
Vice President - Electric Operations
300 Erie Boulevard West
Syracuse, New York 13202

Gentlemen:

Your letter dated November 15, 1973, proposed changes to the Technical Specifications of Provisional Operating License No. DPR-17 for the Nine Mile Point Nuclear Station Unit 1 (NMP-1). The proposed change redefines the conditions for operability of the APRM rod block system to make the requirements consistent with the assumptions used in the rod withdrawal reanalysis and also revises the maximum control rod worth and scram insertion times to make them consistent with the generic reanalysis of the control rod drop accident (RDA).

The proposed change involving the conditions which define operability of the APRM rod block was initiated as a result of the reanalysis of the rod withdrawal transient that was made in association with the present refueling program using 8 x 8 reload fuel. The proposed changes in maximum worth of an in-sequence control rod and in the specified control rod scram insertion times were initiated to bring these limits into conformance with requirements based on the generic reanalysis of the RDA submitted by General Electric (GE) in Topical Report NEDO-10527 "Rod Drop Accident Analysis for Large Boiling Water Reactors," issued March 1972. GE has applied the information and techniques developed in that report to provide a technical basis and Technical Specifications for operating BWR plants including the NMP-1 reactor class.

We have reviewed the RDA analysis you submitted and compared it with the GE analysis and find that the analyses are consistent. During our review we informed your staff that certain modifications were necessary to your proposed Technical Specifications and associated bases to make them conform with our requirements. These modifications have been made.

We have concluded that the proposed changes to the Technical Specifications, as modified, do not present a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered. A copy of our Safety Evaluation regarding these proposed changes is enclosed.

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Accordingly, Amendment No. 4 to Provisional Operating License No. DPR-17 is enclosed revising the Technical Specifications thereto to authorize the requested changes, as modified. A copy of a notice which is being forwarded to the Office of the Federal Register for publication relating to this action also is enclosed for your information.

Sincerely,

Original signed by:
Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosures:
1. Safety Evaluation
2. Amendment No. 4 to License No. DPR-17
3. Federal Register Notice

cc w/encls:
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Mr. Robert P. Jones, Supervisor
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Oswego City Library

cc w/encls and NMP filing dtd. 11/15/73:
Dr. William Seymour
Staff Coordinator
New York State Atomic Energy Council
New York State Department of Commerce
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Mr. Paul Arbesman
Environmental Protection Agency
26 Federal Plaza
New York, New York 10007

Licensee's cy mailed 7/19/74 Rdg

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SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

SUPPORTING AMENDMENT NO. 4 TO LICENSE NO. DPR-17

(CHANGE NO. 12 TO APPENDIX A OF TECHNICAL SPECIFICATIONS)

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT UNIT 1

DOCKET NO. 50-220

INTRODUCTION

By letter dated November 15, 1973, the Niagara Mohawk Power Corporation proposed changes to the Technical Specifications of Provisional Operating License No. DPR-17 for the Nine Mile Point Nuclear Station Unit 1 (NMP-1). The proposed changes consist of a revision of the maximum in-sequence control rod worth, additional restrictions to the APRM rod block system requirements and a revision to the control rod scram insertion times. These changes result from reanalysis of the postulated events of rod withdrawal transient and rod drop accident.

EVALUATION

Rod Drop Accident

The change proposed by Niagara Mohawk concerning the rod drop accident was modified by the addition of restrictions associated with maintaining an operable Rod Worth Minimizer during reactor startup. The proposed changes and our modifications are based on the analytical models and techniques developed on a generic basis by the General Electric Company (GE) and presented in references (1), (2) and (3). The information and techniques developed by GE were used by Niagara Mohawk for application to NMP-1, as described in reference (4) and we compared this with reference (5). We have reviewed reference (5) which was submitted on the Monticello Nuclear Generating Plant Docket (50-263) on October 4, 1973, concluded that it was fully appropriate for the NMP-1 reactor, and used it in our review of the proposed changes. The changes, as authorized, result in a reduction in maximum allowable in-sequence control rod reactivity worth from 2.5% to 1.3% delta k/k, and increase the assurance that a control rod is not in an out-of-sequence position during low power operation.

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The rod drop accident is one of the design basis accidents for boiling water reactors. For calculational purposes, it is assumed that a control rod blade separates from its drive, lodges in the core with the drive withdrawn and drops at the time which causes the most serious power excursion due to rapid reactivity insertion. The consequences of this accident are evaluated by determining the energy input to the fuel assuming that the reactivity worth of the dropped rod is the maximum which could occur. The maximum acceptable energy in the fuel is limited such that, in the event of fuel cladding failure, the energy input into the coolant will not result in a pressure pulse which might damage the core geometry or the reactor pressure vessel.

The analytical methods used by GE to evaluate the consequences of the rod drop accident have been reviewed by the staff and independent calculations have been performed by Brookhaven National Laboratory which show reasonable agreement with GE results. Based on these reviews, it is concluded that the analytical methods used by GE are acceptable.

Application of the GE analytical methods to operating reactors requires that the input parameters conservatively represent the reactor core over a broad range of operating conditions. The proposed changes to the Technical Specifications include, in the bases, a set of boundary conditions which are used to calculate the maximum allowable reactivity worth of control rod. It is not expected that these boundary conditions will be exceeded for reactor cores of current design. The boundary conditions include a maximum inter-assembly local power peaking factor, an end-of-cycle delayed neutron fraction, a beginning of life Doppler reactivity feedback, the technical specification control rod scram insertion rate, a control rod drop velocity of 3.11 ft/sec, and specified accident and scram reactivity shape functions. The rod drop velocity of 3.11 ft/sec is based on tests with a "worst-case" rod built with maximum clearances and features known to contribute to the high rod drop velocities. The difference between the mean rod drop velocity and the 99.9% confidence limit for a group of production rods was added to the mean velocity obtained for the "worst-case" control rod. We have added the value of 0.005 for the end-of-cycle delayed neutron fraction to further define the boundary assumptions. In addition, we have added a statement to the bases that each reload core must be analyzed to show conformance to the bounding assumption. The peak fuel enthalpy resulting from an in-sequence rod drop accident within the above boundary conditions is calculated not to exceed 280 cal/gm, which is acceptably below the peak fuel enthalpy at which prompt fuel dispersal would occur based on the SPERT tests. Based on the above, the resultant maximum allowable in-sequence rod worth of 1.3% delta k/k is acceptable.

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Separate consideration is being given to the potentially adverse effect on the rod drop accident due to the possible presence of inverted poison tubes in the NMP-1 control blades. In our letter dated April 8, 1974, regarding this control rod manufacturing defect, Niagara Mohawk was required to increase the calculated reactivity worth of control blades by 0.2 percent delta k/k, the maximum possible increase assuming full poison settling in the most unfavorable configuration. This requirement, in conjunction with the new Technical Specifications, will effectively limit the allowable in-sequence rod reactivity worth to 1.1% delta k/k until inspections determine the actual extent of inverted poison tubes.

If a control rod is withdrawn out-of-sequence, a rod worth of greater than 1.3% delta k/k could result. In the event of a rod drop accident associated with such an out-of-sequence rod, the peak fuel enthalpy could exceed 280 cal/gm. The rod worth minimizer (RWM) is designed as an operator aid to prevent an out-of-sequence rod withdrawal. Current Technical Specifications allow the RWM to be bypassed if it is inoperable during a reactor startup provided that a second operator is assigned to monitor the rod withdrawal sequence. To increase the control on RWM availability during reactor startups, the technical specification is being changed to require that the RWM be operable for the withdrawal of a significant number of control rods. The effective date of this change concerning RWM operability is being deferred until November 1, 1974, to allow any necessary upgrading on the RWM to be accomplished.

The proposed change in control rod scram times reflect the revised assumptions used in the rod drop accident analysis to be consistent with the generic basis. This change updates the NMP-1 Technical Specifications to those of other GE boiling water reactors. The requested change by Niagara Mohawk included a proposal for longer times (i.e., slower scrams) for scram insertion at reactor pressures below 950 psig. Specification of these longer times for the first 20% of the scram stroke is not necessary because the scram time measurements should be made at reactor pressures above 950 psig which correspond to system operating conditions. For this reason we have modified the proposed change to not include the longer times.

Rod Withdrawal Transient

The Average Power Range Monitoring (APRM) system provides protection against fuel damage in the core in the event of an inadvertent rod withdrawal transient. This transient analysis is made on the assumption that the control rod of maximum worth is fully inserted while the adjacent controls rods are in a withdrawn position such that the full reactor design power and associated design minimum critical heat flux ratio (MCHFR) limit of 1.9 exists near the inserted rod. This maximum worth rod is then inadvertently

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withdrawn until rod block occurs assuming the worst allowable APRM bypass condition. Thus, the APRM rod block system provides local protection of the core fuel by limiting control rod withdrawal so that MCHFR is maintained above 1.0.

The APRM rod block upscale setpoint and the required minimum configuration of signals from the Local Power Range Monitors (LPRM) are determined by the rod withdrawal analysis to assure that no fuel damage (i.e., a MCHFR below unity) occurs as a result of the transient. The present analysis is based on an upscale trip setpoint of the APRM rod block at 106% of the initial level of APRM channel reading. This upscale setpoint was recently reduced to 105% as described in reference (6). The authorized reduction provides additional margin to that shown by the licensee's analysis in reference (4).

The minimum configuration of LPRM inputs is defined for APRM operability. This definition is developed as a result of the rod withdrawal transient analysis and is implemented by the proposed changes to the Technical Specifications.

CONCLUSION

As discussed above, the proposed changes consisting of a reduction in the maximum in-sequence control rod reactivity worth, additional restrictions to the APRM rod block system requirements, and a reduction in the required control rod scram insertion times for the initial 50% of travel, individually and collectively, serve to enhance the safety of operation of the NMP-1 reactor. Therefore, the staff concludes 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; and (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner.

/S/

C. J. DeBevec
Operating Reactors Branch #2
Directorate of Licensing

/S/

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing

Date: JUL 16 1974

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REFERENCES

(1) Paone, C. J., Stirn, R. C., and Wooley, J. A., "Rod Drop Accident Analysis for Large Boiling Water Reactors", NEDO-10527, March 1972.

(2) Stirn, R. C., Paone, C. J., and Young, R. M., "Rod Drop Accident Analysis for Large BWR's," Supplement 1 - NEDO-10527, July 1972.

(3) Stirn, R. C., Raone, C. J., and Haun, J. M., "Rod Drop Accident Analysis for Large Boiling Water Reactors Addendum No. 2 Exposed Cores," Supplement 2 - NEDO-10527, January 1973.

(4) Letter dated November 15, 1973, from R. R. Schneider, Niagara Mohawk Power Corporation, to A. Giambusso, USAEC, with Attachments A and B.

(5) Report entitled "Technical Basis for Changes to Allowable Rod Worth Specified in Technical Specification 3.3.B.3" transmitted by letter from L. O. Mayer to J. F. O'Leary (USAEC) dated October 4, 1973 (Northern States Power, Docket No. 50-263).

(6) Change No. 11 for Nine Mile Point Unit 1, Docket No. 50-220, License DPR-17, Letter from D. J. Skovholt to Philip D. Raymond, Niagara Mohawk Power Corporation, dated April 10, 1974.

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UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION UNIT 1

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 4
License No. DPR-17

1. The Atomic Energy Commission (the Commission) has found that:

   A. The application for amendment by the Niagara Mohawk Power Corporation (the licensee) dated November 15, 1973, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended, and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. 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

   C. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

2. Accordingly, paragraph 3.B of Provisional Operating License No. DPR-17 is hereby amended to read as follows:

   "B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, attached to Provisional Operating License No. DPR-17 are revised as indicated in the attachment to this license amendment. The Technical Specifications, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised."

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

FOR THE ATOMIC ENERGY COMMISSION

Karl R. Goller, Assistant Director
for Operating Reactors
Directorate of Licensing

Attachment:
Change No. 12 to Appendix A
Technical Specifications

Date of Issuance: JUL 1 6 1974

ATTACHMENT TO LICENSE AMENDMENT NO. 4

CHANGE NO. 12 TO APPENDIX A OF TECHNICAL SPECIFICATIONS

PROVISIONAL OPERATING LICENSE NO. DPR-17

1. Replace Limiting Condition for Operation 3.1.1.b(3), page 23, with the following:

   (3) (a) Control rod withdrawal sequences shall be established so that maximum reactivity that could be added by dropout of any increment of any one control blade would not make the core more than 0.013 delta k supercritical.

   (b) Whenever the reactor is in the startup or run mode below 10% rated thermal power, no control rods shall be moved unless the rod worth minimizer is operable or a second independent operator or engineer verifies that the operator at the reactor console is following the control rod program. After November 1, 1974, 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.

2. Replace Surveillance Requirement 4.1.1.b(3), page 23, with the following:

   (3) (a) To consider the rod worth minimizer operable, the following steps must be performed:

   (i) The control rod withdrawal sequence for the rod worth minimizer computer shall be verified as correct.

   (ii) The rod worth minimizer computer on-line diagnostic test shall be successfully completed.

   (iii) Proper annunciation of the select error of at least one out-of-sequence control rod in each fully inserted group shall be verified.

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(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 10% 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.

3. Replace Basis b(3), page 23, with the following:

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 10% 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 10% power, even single 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 specified 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.

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a. A startup inter-assembly local power peaking factor of 1.30 or less.(6)

b. An end of cycle delayed neutron fraction of 0.005.

c. A beginning of life Doppler reactivity feedback.

d. The Technical Specification rod scram insertion rate.

e. The maximum possible rod drop velocity (3.11 ft/sec).

f. The design accident and scram reactivity shape function.

g. The moderator temperature at which criticality occurs.

It is recognized that these bounds are conservative with respect to expected operating conditions. If any one of the above conditions is not satisfied, a more detailed calculation will be done to show compliance with the 280 cal/gm design limit.

In most cases the worth of in-sequence rods or rod segments will be substantially less than 0.013 delta k. Further, the addition of 0.013 delta k worth of reactivity as a result of a rod drop in conjunction with the actual values of the other important accident analysis parameters described above would most likely result in a peak fuel enthalpy substantially less than the 280 cal/gm design limit. However, the 0.013 delta k limit is applied in order to allow room for future reload changes and ease of verification without repetitive Technical Specification changes.

Should a control rod drop accident result in a peak fuel energy content of 280 cal/gm, less than 660 (7 x 7) fuel rods are conservatively estimated to perforate. This would result in offsite doses greater than previously reported in the FSAR, but still well below the guideline values of 10 CFR 100. For 8 x 8 fuel, less than 850 rods are conservatively estimated to perforate, which has nearly the same consequences as for the 7 x 7 fuel case because of the operating rod power differences.

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

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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 10% 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 after November 1, 1974.

4. Include the following as references to Bases at the bottom of page 23:

   (1) Paone, C. J., Stirn, R. C., and Wooley, J/ A., "Rod Drop Accident Analysis for Large Boiling Water Reactors," NEDO-10527, March 1972.

   (2) Stirn, R. C., Paone, C. J., And Young, R. M., "Rod Drop Accident Analysis for Large BWR's," Supplement 1 - NEDO-10527, July 1972.

   (3) Stirn, R. C., Paone, C. J., and Haun, J. M., "Rod Drop Accident Analysis for Large Boiling Water Reactors Addendum No. 2 Exposed Cores," Supplement 2 - NEDO-10527, January 1973.

   (4) Report entitled "Technical Basis for Changes to Allowable Rod Worth Specified in Technical Specification 3.3.B.3." transmitted by letter from L. O. Mayer (NSP) to J. F. O'Leary (USAEC) dated October 4, 1973.

   (5) Letter, R. R. Schneider, Niagara Mohawk Power Corporation to A. Giambusso, USAEC, dated November 15, 1973.

   (6) To include the power spike effect caused by gaps between fuel pellets.

5. Replace Limiting Condition for Operations 3.1.1.c(1) and 3.1.1.c(2), page 24, with the following:

   c. Scram Insertion Times

      (1) The average scram insertion time of all operable control rods, in the power operation condition, shall be no greater than:

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% Inserted From Fully Withdrawn	Average Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.00
90	5.00

(2) Except as noted in 3.1.1.c.(3), the maximum insertion scram time, in the power operation condition, shall be no greater than:

% Inserted From Fully Withdrawn	Maximum Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.12
90	5.30

6. Replace Basis c, page 24, with the following:

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 are still valid since the use of the revised scram insertion times would result in greater margins to safety valves lifting.

The insertion times previously selected were based on the large number of actual scrams of prototype control rod drive mechanisms as discussed in Section IV-B.6.3*. Rapid control rod insertion following a demand to scram will terminate Station transients before any possibility of damage to the core is approached. The primary consideration in setting scram time is to permit rapid termination of steam generation following an isolation transient (i.e., main-steam-line closure or turbine trip without bypass) such that

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operation of solenoid-actuated relief valves will prevent the safety valves from lifting. Analyses presented in Appendix E-I*, the Second Supplement, and the Technical Supplement to Petition to Increase Power Level were based on times which are slower than the proposed revised times.

The scram times generated at each refueling outage when compared to previous scram times demonstrate that the control rod drive scram function has not deteriorated.

*FSAR

7. Make the additions on the pages identified as follows:

Page 94 - Bases

In the third paragraph an asterisk is used to refer to the FSAR for the bases for allowable LPRM bypass conditions. Add to that reference, "Letter, R. R. Schneider, Niagara Mohawk Power Corporation, to A. Giambusso, USAEC, dated November 15, 1973".

Page 99 - Notes for Tables 3.6.2a and 4.6.2a:

a. In note (e), add the following to the last sentence "...provided that the APRM in the other instrument channel in the same core quadrant is not bypassed."

b. Add to note (e), the following new sentence: "A Travelling In-Core Probe (TIP) chamber may be used as a substitute APRM input if the TIP is positioned in close proximity to the failed LPRM it is replacing."

Page 108 - Notes for Tables 3.6.2g and 4.6.2g:

a. In note (c) add the following to the first sentence "...provided that the APRM in the other instrument channel in the same core quadrant is not bypassed."

b. Add to note (c), the following new sentence: "In the Run mode of operation, bypass of two chambers from one radial core location in any one APRM shall cause that APRM to be considered inoperative."

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c. Add to note (c), the following new sentence: "A Travelling In-Core Probe (TIP) chamber may be used as a substitute APRM input if the TIP is positioned in close proximity to the failed LPRM it is replacing."

d. Add to note (c), the following new sentence: "If one APRM in a quadrant is bypassed and meets all requirements for operability with the exception of the requirement of at least one operable chamber at each radial location, it may be returned to service and the other APRM in that quadrant may be removed from service for test and/or calibration only if no control rod is withdrawn during the calibration and/or test."

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UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-220

NIAGARA MOHAWK POWER CORPORATION

NOTICE OF ISSUANCE OF PROVISIONAL OPERATING LICENSE AMENDMENT

Notice is hereby given that the U. S. Atomic Energy Commission (the Commission) has issued Amendment No. 4 to Provisional Operating License No. DPR-17 issued to the Niagara Mohawk Power Corporation which revised the Technical Specifications for operation of the Nine Mile Point Nuclear Station Unit 1 located in Oswego County, New York.

The amendment revised the Technical Specifications to redefine the conditions for operability of the Average Power Range Monitoring rod block system to make the requirements consistent with the assumptions used in the rod withdrawal reanalysis and also revised the maximum control rod worth and scram insertion times to make them consistent with the generic reanalysis of the control rod drop accident.

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, and 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.

For further details with respect to this action, see (1) the application for amendment dated November 15, 1973, (2) Amendment No. 4 to License No. DPR-17, with an attachment, 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 at 120 East Second Street, Oswego, New York 13126.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Atomic Energy Commission, Washington, D. C. 20545, Attention: Deputy Director for Reactor Projects, Directorate of Licensing - Regulation.

Dated at Bethesda, Maryland, this 16th day of July 1974

FOR THE ATOMIC ENERGY COMMISSION

Original signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

→ Boyd

June 21, 1974

Sounds OK on quick reading. Appears to be a close one tho.

Note to: Ed Case

Our discussions on the relationship between significant new safety information and significant hazards considerations always end up with some amount of uncertainty. The attached change for NMP-1 is an excellent illustration of a case involving the former, but not the latter. Angie and I have both reviewed this and have concluded there is no need for prior notice. I'd like you to give it a quick look; it will crystalize your thinking on the subject.

When you're finished, either send it back to me, or on to OGC.

EC 6/21

Roger S. Boyd, ADDRP

Attachment:
Proposed ltr to Niagara
Mohawk re tech spec chng