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APR 10 1974

Docket No. 50-220

Niagara Mohawk Power Corporation
ATTN: Mr. Philip D. Raymond
Vice President - Engineering
300 Erie Boulevard West
Syracuse, New York 13202

Change No. 11
License No. DPR-17

Gentlemen:

Your letter dated September 14, 1973, requested authorization to refuel and operate the Nine Mile Point Nuclear Station Unit 1 reactor (NMP-1) with up to 120 fuel assemblies of the 8 x 8 design. Your letters dated October 15, 1973, January 15 and 22, 1974, and February 19, 1974, provided supplemental information in support of your request, including proposed changes to the Technical Specifications of Provisional Operating License No. DPR-17.

The use of 8 x 8 fuel in reloads has been reviewed on a generic basis by the Licensing staff and the Advisory Committee on Reactor Safeguards (ACRS). The reports based on these reviews were transmitted to you by letters dated February 11 and 20, 1974. The staff Safety Evaluation for the use of 8 x 8 fuel assemblies in the Nine Mile Point Reactor, including the proposed changes to the Technical Specifications, is enclosed for your information. Based on these reviews, we have concluded that the health and safety of the public will not be endangered by the proposed refueling and subsequent operation with the 8 x 8 fuel and with the proposed changes to the Technical Specifications. Therefore, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications of License No. DPR-17 are hereby changed as shown by the enclosed Change No. 11, effective immediately.

As required by 10 CFR Part 2, the enclosed notice relating to the issuance of this change is being filed with the Office of the Federal Register for publication. A copy of the Atomic Safety and Licensing Board's Memorandum and Order dated April 8, 1974, ruling on the New York Atomic Energy Council's request for leave to intervene as an interested State, also is enclosed for your information.

Informed G. Rhode of Niagara by phone 4-10-74 @ 4:30 pm C.J.D.

Sincerely,

/S/

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

CP+8

enclosures:
see next page

L:ORB #2	L:ORB #2	L	OGC	L:RS	L:OR
CDeBevec	RMDiggs	DLZiemann	JGallo	VStello	DJS
4/10/74	4/10/74	4/10/74	4/10/74	4/10/74	

Form AEC-318 (Rev. 9-53)
AECM 0240
GPO c43-16-81465-1 445-678

Enclosures:
1. Change No. 11
2. Safety Evaluation
3. Federal Register Notice
4. Board's Memo and Order of 4/8/74

cc w/enclosures:
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Commerce and Counsel to the
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Oswego City Library

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Staff Coordinator
New York State Atomic Energy Council
New York State Department of Commerce
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(w/cy of NMP filing dtd. 2/19/74)

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ATTN: Joan Sause
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Washington, D. C. 20460
(w/cy of NMP filing dtd. 2/19/74)

Mr. Paul Arbesman
Environmental Protection Agency
26 Federal Plaza
New York, New York 10007
(w/ cy of NMP filing dtd 2/19/74)

OFFICE ▶						
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ATTACHMENT A

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-17

DOCKET NO. 50-220

CHANGE NO. 11 TO TECHNICAL SPECIFICATIONS

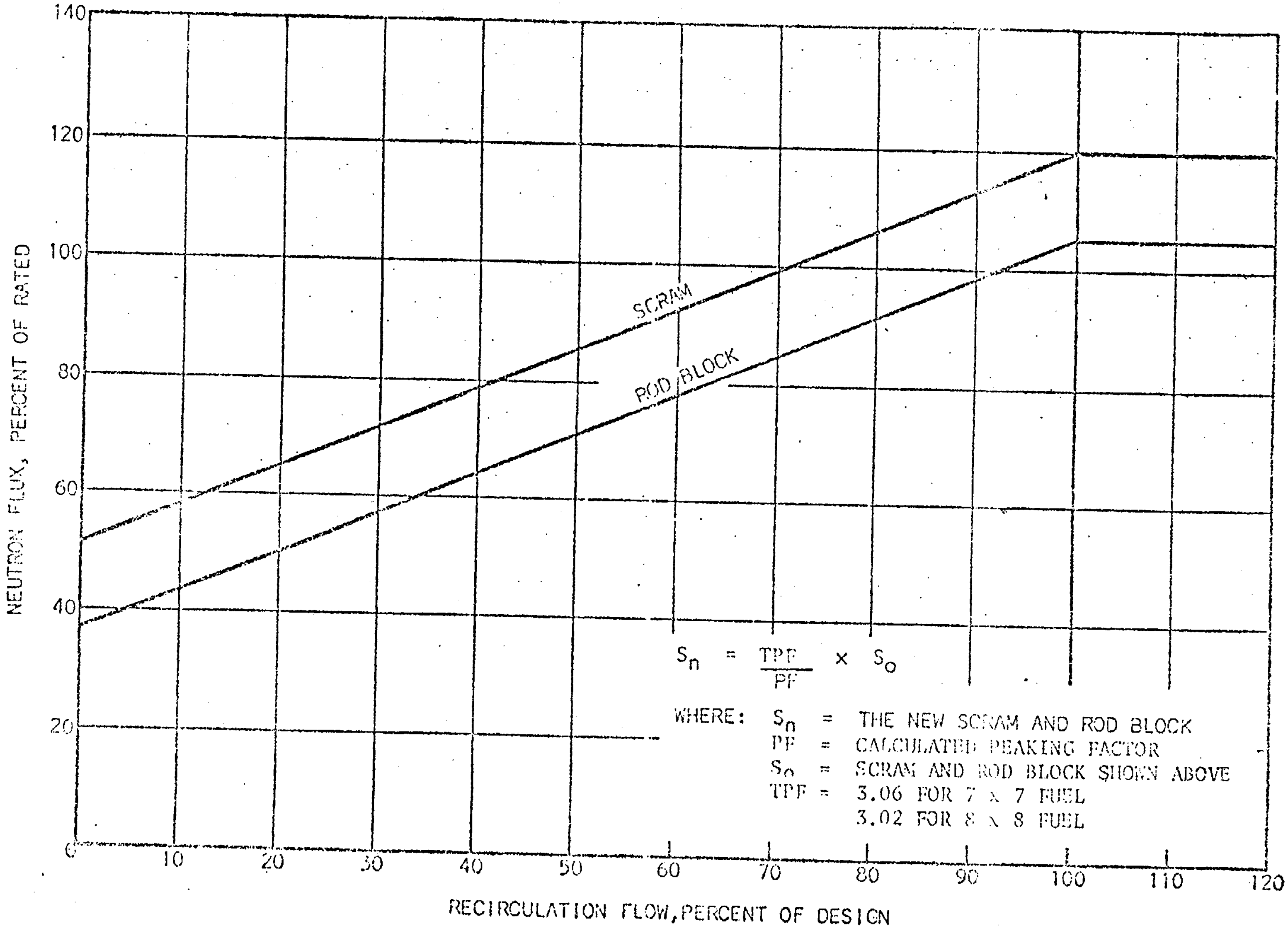
Item	Location	Change
Basis Statement 2.1.1	page 7	Insert the following words after the number 3.06 in lines 3, 10, and 11 of the last paragraph "for 7 x 7 fuel and 3.02 for 8 x 8 fuel".
Safety Limit: Figure 2.1.1 Revised	page 10	Replace the number 3.06 under notes and in the equations with the letters "TPF" and add the following: "TPF = 3.06 for 7 x 7 fuel 3.02 for 8 x 8 fuel"
Basis Statements for 2.1.2 and 2.1.2.a	page 11	Add a superscript 11 to 6, 7, 9, and 10 on line 4 and on last line of paragraph on page. At bottom of page add as Reference 11, Letter, Philip D. Raymond, Vice President - Engineering, Niagara Mohawk Power Corporation, to A. Giambusso, Deputy Director for Reactor Projects, USAEC, dated October 15, 1973.
Basis Statement for 2.1.2.a	page 12	Replace the first sentence in the last paragraph with the following: "The thermal hydraulic safety limit of Figure 2.1.1 Revised was based on a total peaking factor of 3.06 for 7 x 7 fuel design and 3.02 for 8 x 8 fuel design and an adjustment is required in the unusual event of higher peaking factors".
Basis Statement for 2.1.2.f	page 14	Change the figure on the third line from 106 to 105.

Item	Location	Change
Basis Statement for 2.1.2g-h	page 14	Change the last sentence of the first paragraph by adding as a reference, Letter, Philip D. Raymond, Vice President - Engineering, Niagara Mohawk Power Corporation, to A. Giambusso, Deputy Director for Reactor Projects, USAEC, dated October 15, 1973.
Basis Statement for 2.1.2.i	page 15	In the parenthetical reference add as a reference, Letter, Philip D. Raymond, Vice President - Engineering, Niagara Mohawk Power Corporation, to A. Giambusso, Deputy Director for Reactor Projects, USAEC, dated October 15, 1973.
Limiting Safety System Setting: Figure 2.1.2 Revised	page 16	Replace Figure 2.1.2 Revised with the attached Figure 2.1.2 Revised.
Limiting Condition for Operation 3.1.7.a	page 37a	Add the attached Figure 3.1.7e to the Technical Specification. Under the Limiting Condition for Operation 3.1.7.a, add Figure 3.1.7e to the list of figures at the end of the last sentence in paragraph a.
Limiting Condition for Operation 3.1.7.b	page 37a	Replace the notes to the equation in Limiting Condition for Operation 3.1.7.b with the following: "LHGR = Design LHGR = 17.5 kW/ft for 7 x 7 fuel or 13.4 kW/ft for 8 x 8 fuel $(\text{Delta } p/p)_{MAX}$ = Maximum power spiking penalty = 0.040 for 7 x 7 fuel or 0.027 for 8 x 8 fuel. LT = Total Core length = 12 ft. L = Axial position above bottom of core"
Basis Statement for 3.1.7.a	page 37a	Add Figure 3.1.7.e to the list of figures in the third paragraph of the Bases 3.1.7.a.

Item	Location	Change
Basis Statement for 3.2.9.a	page 58	In the second paragraph add as a reference, Letter, Philip D. Raymond, Vice President - Engineering, Niagara Mohawk Power Corporation, to A. Giambusso, Deputy Director for Reactor Projects, USAEC, dated October 15, 1973, at the end of the sentence.

FIGURE 2.1.2 REVISED

FLOW BIASED SCRAM AND APRM ROD BLOCK



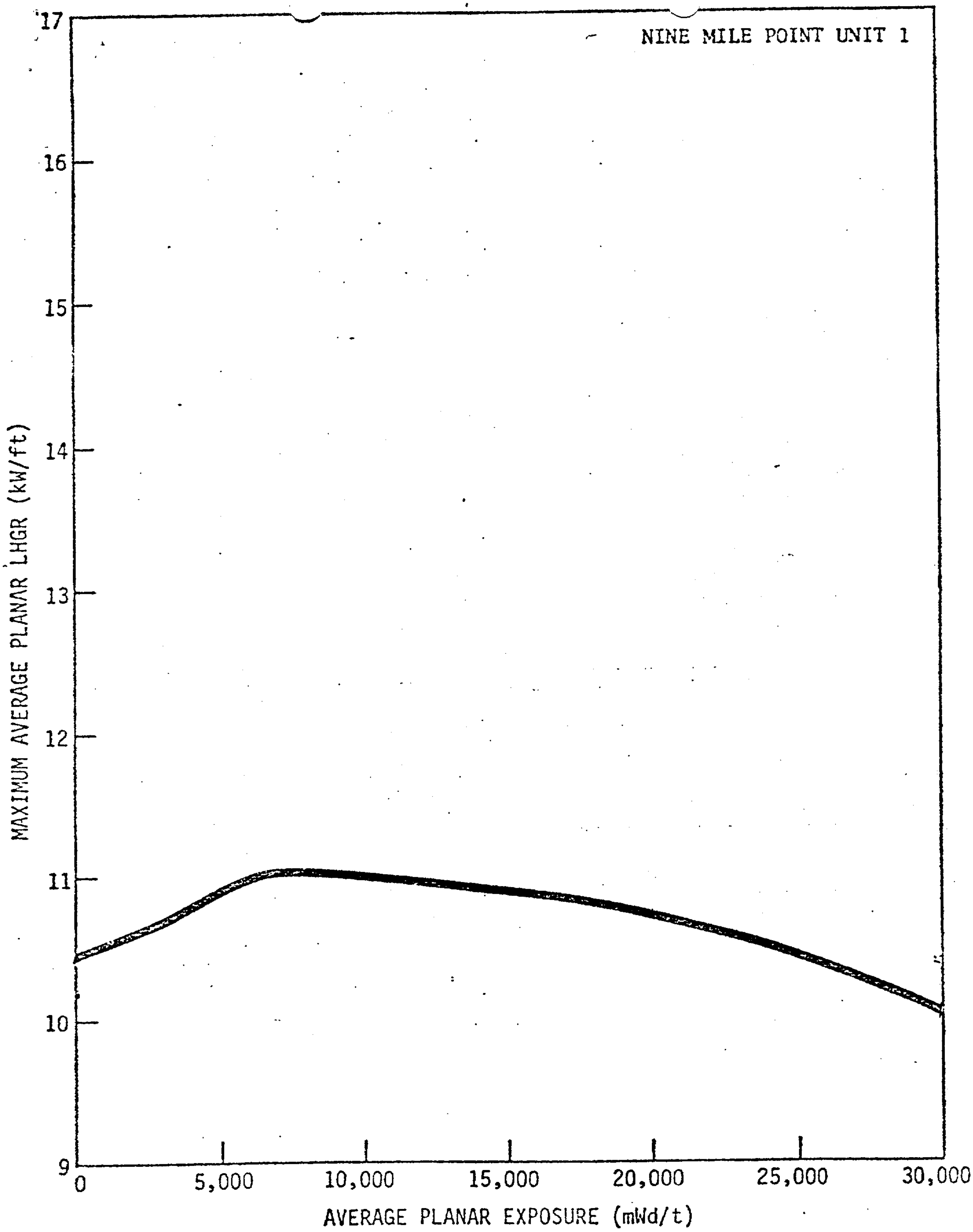


FIGURE 3.1.7.e MAXIMUM ALLOWABLE AVERAGE PLANAR LHGR
APPLICABLE TO FUEL TYPE 5 and 6

UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-220

PROPOSED REFUELING WITH 8 x 8 FUEL

NINE MILE POINT NUCLEAR STATION UNIT 1

INTRODUCTION

By letter dated September 14, 1973, Niagara Mohawk Power Corporation requested authorization to refuel and operate the Nine Mile Point Nuclear Station Unit 1 (NMP-1) with up to 120 fuel assemblies of the 8 x 8 design. This refueling is scheduled for the spring of 1974 and comprises the second major refueling of the NMP-1 reactor core. Additional information in support of this request was provided by Niagara Mohawk by letters dated October 15, 1973, January 15 and 22, 1974, and February 19, 1974. These letters, respectively, provided information regarding the effect of the 8 x 8 reload fuel on anticipated transients and postulated accidents, information in response to our questions, information regarding the effects of fuel densification on the proposed 8 x 8 reload fuel, and the remaining additional information in response to our requests. The proposed action also includes requests for changes to the technical specifications related to fuel densification considerations and to the rod block monitor.

The safety analysis of the refueled core with the proposed 8 x 8 reload fuel submitted by Niagara Mohawk includes consideration of the effect of the 8 x 8 fuel on previously analyzed conditions during normal operation, operational transients, and postulated accidents. Included also is consideration of the applicability of existing technical specification limits and an evaluation of proposed changes to these limits. The 8 x 8 reload fuel consists of two different enrichments which are identified as Type 5 and Type 6 fuel assemblies. The analyses included consideration of these reload fuel types in the reactor core in combination with initial fuel and previous reload fuel.

The neutronic, thermal-hydraulic, and mechanical acceptability of the 8 x 8 fuel assembly design during normal operation, operational transients, and postulated accidents was evaluated by the Regulatory staff in a separate report(1). This staff report includes an evaluation of the safety of up to a full core loading of 8 x 8 fuel assemblies as compared with a core loading of 7 x 7 fuel assemblies. The use of 8 x 8 fuel for reload cores was also reviewed by the Advisory Committee on Reactor Safeguards and discussed in its report dated February 12, 1974.

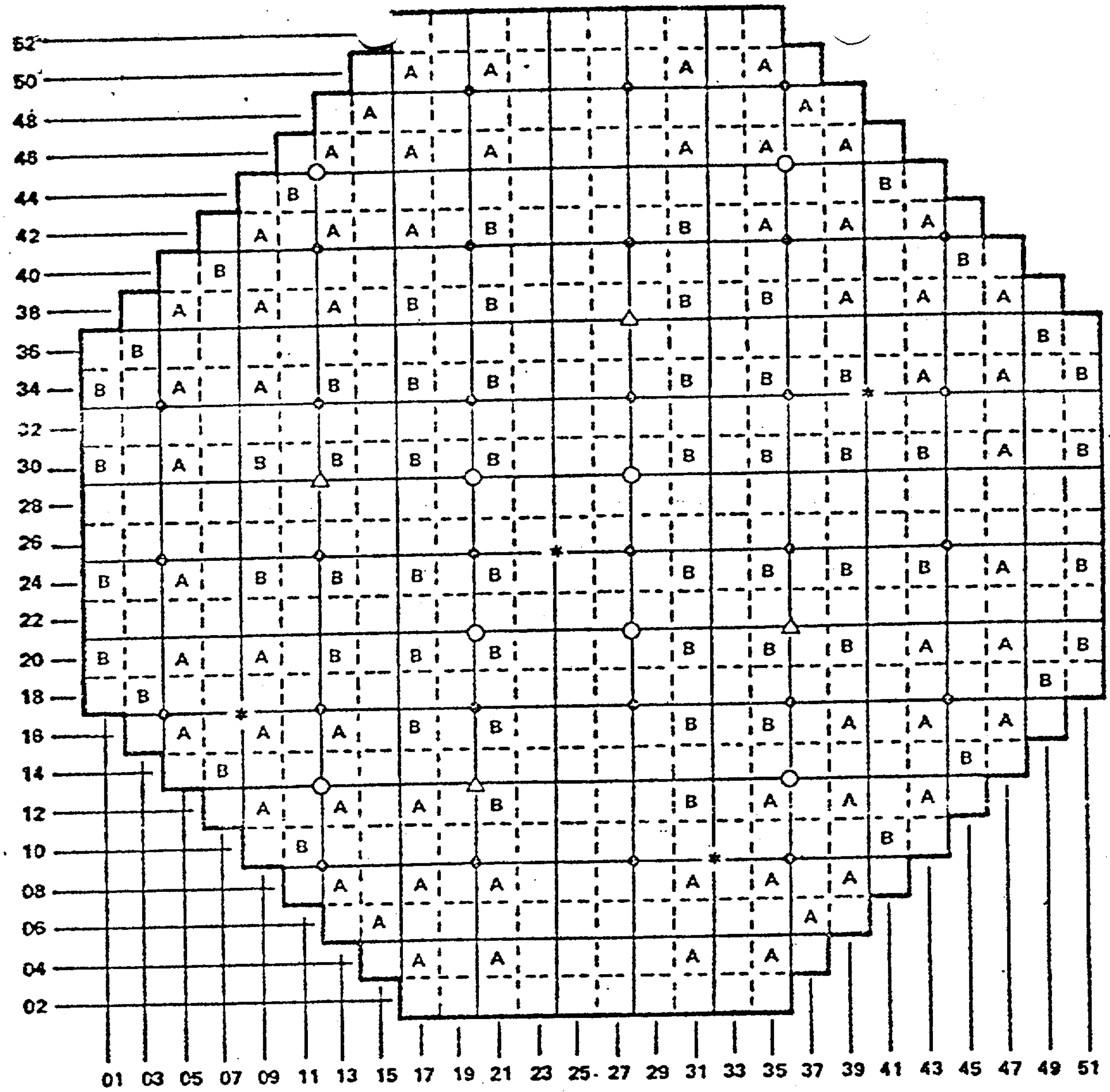
EVALUATION

The 8 x 8 reload fuel has been designed to be compatible with and closely match the mechanical, nuclear, and thermal-hydraulic characteristics of the NMP-1 initial core and the previous reload 7 x 7 fuel. The reference core is based on a scatter pattern of reloading the 8 x 8 fuel as shown by Figure 1. No significant fuel loading asymmetries will exist. As shown, the reloading is in a one-in-four arrangement involving either a Type 5 or Type 6 fuel assembly. The only difference between a Type 5 and Type 6 fuel assembly is the enrichment with Type 5 containing an average enrichment of 2.62 weight percent of U-235 and Type 6 containing 2.50 weight percent of U-235. These fuel types and loading fall within the scope of the staff report(1) on the 8 x 8 fuel assembly. The thermal-hydraulic limiting conditions of operation and the response of the coolant circulation system is consistent with that used in the staff report. The methods of analysis used by the licensee are identical to the methods approved by the staff. Therefore, the evaluations and conclusions of the staff report with respect to normal operations, abnormal operational transients, and accidents are fully applicable to NMP-1.

The Regulatory staff's review(1) of the mechanical design of the 8 x 8 reload fuel concludes that the background of experience compiled by the General Electric Company is sufficient to enable GE to design fuel rods of new design with confidence in their durability. The NMP-1 8 x 8 fuel assemblies are of similar design and material as the 7 x 7 fuel assemblies which have successfully been operated at NMP-1 for over 4 years. Both the 8 x 8 and 7 x 7 assemblies will operate at the same pressure and temperature and the fluid velocity and quality will be nearly identical and, therefore, the 8 x 8 fuel assemblies are expected to exhibit the same operational characteristics as the previously operated 7 x 7 assemblies.

Accident induced loads and stresses have been calculated for both the 7 x 7 and 8 x 8 assemblies using the same methods. The limiting accident loads result from a steam line break. The pressure differences following a steam line break are less than 10% greater than normal operating pressure differences. As in normal operation, the pressure differences in an 8 x 8 assembly following a steam line break are 5 to 10% greater than in a 7 x 7 assembly. The loads following a steam line break are well below the allowable loads.

Based upon the above, the staff concludes that the mechanical design of the NMP-1 8 x 8 reload fuel is adequate to assure the mechanical integrity of the fuel assemblies. Additional assurance of acceptable fuel performance of the new fuel design is provided by the radiological surveillance maintained on the reactor primary coolant and off-gas to provide an early indication of incipient fuel failure caused by mechanical deterioration of the fuel assemblies.



- o LPRM LOCATION
- ○ IRM LOCATION
- △ SRM LOCATION
- * SOURCE LOCATION
- A RELOAD Type 5
- B RELOAD Type 6

NOTE: ASSEMBLIES WITH A FACE ON THE CORE PERIPHERY HAVE FLOW RESTRICTIVE ORIFICING

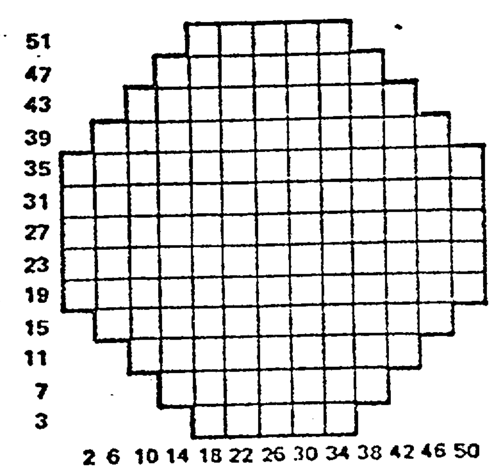


Figure 1. Design Reference Core Loading

We have also reviewed the nuclear characteristics of the 8 x 8 reload fuel. The fuel is identical to that which was evaluated in the Regulatory staff's evaluation(1) of 8 x 8 fuel elements, including both types of fuel enrichment. Based on the staff evaluation as reported(1) we conclude that a mixed 8 x 8 and 7 x 7 core will be similar neutronically, to a 7 x 7 core and that the nuclear design is acceptable.

The staff evaluation(1) of the expected thermal-hydraulic performance uses identical fuel damage limits and thermal-hydraulic criteria to evaluate both the 8 x 8 and 7 x 7 assemblies. The results of this evaluation show that the 8 x 8 assembly minimum critical heat flux ratio (MCHFR) is expected to be 11% greater than the MCHFR for a 7 x 7 assembly operating under similar conditions of flux peaking. Additionally, the 8 x 8 fuel assemblies operating at their design value provide 20% greater margin to the 1% cladding strain criteria than the 7 x 7 assemblies and the margin of design linear heat generation rate to pellet center line melting is 17% higher for 8 x 8 assemblies than for 7 x 7 assemblies. The staff has reviewed the thermal-hydraulic differences between the 7 x 7 and 8 x 8 assemblies involving a modified flow geometry and the introduction of an unfueled rod. The modified flow geometry will provide a more balanced subchannel flow in the 8 x 8 assembly than in the 7 x 7 bundle and, therefore, we conclude that the thermal performance is improved. The effect of the unheated rod has been previously reviewed(2) and the staff concluded that the effect of the unheated rod is not significant.

Based upon the above considerations the staff concludes that the thermal-hydraulic performance of the NMP-1 8 x 8 reload fuel is acceptable and will provide an increased margin of safety as compared with the previously operated 7 x 7 assemblies.

A. Proposed Changes to Technical Specifications

Although the performance characteristics of the 8 x 8 reload fuel are similar to previously authorized loadings, certain changes to the Technical Specifications are necessary to accommodate this fuel. These changes consist of specifying a total peaking factor of 3.02 for the 8 x 8 fuel (as compared with a factor of 3.06 for the 7 x 7 fuel), reducing the upscale set point of the APRM rod block from the present 106 percent to 105 percent of rated power, and incorporating the appropriate limits on average planar and local linear heat generation rates to reflect fuel densification effects on the 8 x 8 fuel.

The proposed change in total peaking factor recognizes that different total peaking factors are used for the 7 x 7 and 8 x 8 fuel assembly designs. The difference in total peaking factor results from the change in the maximum local peaking factor of 1.22 for the 8 x 8 fuel design compared with 1.30 for the 7 x 7 fuel. The staff report[1] notes the change in local peaking factor and concludes that the nuclear design of the 8 x 8 fuel is acceptable. On the basis of the staff report[1], we conclude that the change in peaking factor is acceptable for NMP-1.

The APRM provides protection of the core in the event of an inadvertent withdrawal of a control rod of high reactivity worth from an assumed control rod pattern which is not normally used, but which maximizes control rod reactivity worth for purposes of the analysis. The APRM rod block provides local protection of the core fuel by limiting control rod withdrawal so that the minimum critical heat flux ratio (MCHFR) is maintained above 1.0. At present, the upscale trip set point of the APRM rod block is specified to be 106 percent of the initial level of APRM channel reading. At this setting, the FSAR analysis showed rod withdrawal being blocked when MCHFR was 1.4. As a result of changes in core characteristics during the previous reloads and with the introduction of 8 x 8 fuel, the reanalysis by Niagara Mohawk shows that the upscale trip set point has to be reduced to provide a MCHFR above 1.0. The proposed change would reduce the trip set point to 105 percent of the initial level of APRM channel reading. At this reduced setting and with the reanalysis, the rod withdrawal is shown to be blocked when MCHFR is about 1.15. Although the margin to a MCHFR of 1.0 is reduced, it is still within the range of approved values for other reactors and provides a margin which we consider acceptable. The proposed setting reduction to a more restrictive value is therefore acceptable.

Average planar and local LHGR is a function of the fuel type and is related to fuel densification. Since a new fuel type (the 8 x 8) is being added to the core, new limitations must be incorporated in the Technical Specifications. By letter dated January 22, 1974, the applicant provided proposed maximum average planar LHGR limits applicable to the 8 x 8 fuel. The applicant used the present approved fuel rod thermal performance model, GEGAP-III[3][4], as the basis for the MAPLHGR limits. The cladding collapse model and power spike model used in the applicant's analysis are consistent with the previous basis[3][4]. Therefore, the proposed technical specifications for MAPLHGR and local LHGR, calculated by use of the approved fuel densification model, are acceptable.

B. Abnormal Operational Transients

Abnormal operational transients were discussed in the staff report for 8 x 8 reload(1) and it was concluded that the reload fuel met the applicable criteria. As previously discussed, the mechanical, nuclear, and thermal-hydraulic characteristics of the 7 x 7 and 8 x 8 fuel are similar and will respond to transients similarly. Also the slight reduction in flow in the 8 x 8 assemblies due to slight increase in flow resistance will be offset by an accompanying increase in the 7 x 7 assemblies and the effect on the total core flow will be negligible.

The staff report(1) also concluded that the replacement of the 7 x 7 assemblies with 8 x 8 assemblies will not result in exceeding fuel damage limits during anticipated transients. The licensee has analyzed the events which have limiting MCHFRs including trip of recirculation pumps, a one pump seizure, a continuous withdrawal of a control rod, and misorientation of a reloaded fuel assembly. The results of these analyses show that the fuel damage limits, i.e., a MCHFR of unity and a cladding stain of one percent, are not reached during these transients. On the basis of the above, we have concluded that the NMP-1 reactor, when reloaded with the proposed fuel and operated in accordance with the proposed changes in technical specifications, satisfies the fuel damage criteria for the abnormal operational transients.

C. Accident Analysis

The generic reevaluation of accidents to account for the effects of 8 x 8 fuel was discussed in the staff evaluation(1) and is applicable to NMP-1. That evaluation noted that the plant specific aspects of the review, such as compliance with the Interim Acceptance Criteria for Emergency Core Cooling, including the effects of densification, any necessary revisions to Technical Specifications requirements, and radiological consequences of postulated accidents would be addressed in a separate evaluation for the specific plant. The Technical Specifications changes, including those associated with densification, have been discussed above.

The Regulatory staff has reviewed the analysis of the loss-of-coolant accident on a generic basis and has concluded that the General Electric Evaluation Model (NEDO-10329), as modified by GE in NEDE-10801 to account for differences in geometry and subsequently modified by the staff to account for the effects of fuel densification, is applicable to the evaluation of the Emergency Core Cooling performance with 8 x 8 fuel assemblies in a General Electric boiling water reactor which has jet pumps. As noted in the staff generic report[1], the applicability of the analytical model to non-jet pump plants is under review. Since the NMP-1 reactor is a non-jet pump plant, this review is being made for NMP-1. At the present time, information on 7 x 7 fuel bundle core spray tests with simulated clad swelling indicates inconsistent results with the analytical model. This aspect of our evaluation is not yet complete; however, it is apparent that the use of 8 x 8 fuel is not dependent upon these test results. Therefore, our review of this matter will continue on a generic basis applicable to both 7 x 7 and 8 x 8 fuel types. We plan to complete our review of this generic matter in connection with our evaluation of the NMP-1 conformance with the recent AEC rule on ECCS, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water-Cooled Nuclear Power Reactors," published in the Federal Register on January 4, 1974.

The analytical model for the evaluation of the NMP-1 loss-of-coolant accident for assessing conformance with the Interim Acceptance Criteria for the 8 x 8 fuel is essentially the same as that used for the 7 x 7 fuel. In comparing the analyses, the only changes are those due to fuel geometry and LHGR and these are accounted for in application of the model. The ECCS performance at NMP-1, when operating in accordance with proposed Technical Specifications, has been evaluated to show that the peak clad temperature for both the 7 x 7 initial loading and the 8 x 8 reload fuel remains below 2300°F, and that the metal water reaction for the 8 x 8 fuel assemblies is less than one percent, thereby meeting the requirements of the Interim Acceptance Criteria for Emergency Core Cooling.

The radiological consequence of the postulated accidents is a function of the fission product release, including any change in fission product release because of the use of 8 x 8 fuel. The radiological consequences of a steam line break, fuel handling, control rod drop, and loss-of-coolant accidents were considered. As noted in the staff report on 8 x 8 fuel[1], the steam line break accident is almost entirely dependent on the limits placed on concentration of radioactivity in the primary coolant. These limits are not being modified and, therefore, the radiological consequences remain essentially unchanged. The resulting radiological doses will remain under 10 CFR Part 100 guidelines.

The fuel handling accident is dependent on the damage resulting from dropping an irradiated fuel element on other fuel elements. Since an 8 x 8 fuel assembly is the same size and approximately the same weight as a 7 x 7 assembly, it would impart the same energy to the same number of fuel assemblies as a dropped 7 x 7. Since the 8 x 8 fuel assembly design and fission product inventory are similar to the 7 x 7, the radiological consequences of dropping an assembly onto an 8 x 8 assembly will not be significantly different. The doses from a refueling accident are calculated to be less than 10 CFR Part 100 guidelines. Analyses of the control rod drop accident demonstrate that the dropping of an in-sequence control rod of maximum reactivity worth will not result in a peak fuel pellet enthalpy which exceeds the limit of 280 calories/gram. The number of 8 x 8 rods in the core which would perforate as a result of such an energy deposition is estimated to be higher than the number of 7 x 7 rods which would perforate as a result of a rod drop accident. However, the radiological consequences would be nearly the same because rod power is lower in the 8 x 8 fuel and the fission product inventory no greater than in a 7 x 7 assembly. The design basis loss-of-coolant accident doses are based on a conservatively large fission product inventory release which is independent of the number of perforations which would occur during a LOCA. Therefore, the radiological consequences of the design basis loss-of-coolant accident would also remain unchanged by the use of 8 x 8 fuel assemblies.

CONCLUSION

Based on the above, we have concluded that the health and safety of the public will not be endangered by the proposed reloading and subsequent operation following refueling of NMP-1 with 8 x 8 fuel and with the proposed modifications to the Technical Specifications.

C. J. DeBevec

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

Dennis L. Ziemann

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

Date: April 10, 1974

REFERENCES

1. Technical Report on the General Electric Company 8 x 8 Fuel Assembly, February 5, 1974, Regulatory Staff, U. S. Atomic Energy Commission.

2. Change No. 17 for Oyster Creek, Docket No. 50-219, License DPR-16. Letter from D. J. Skovholt to Ivan Finfrock, Jersey Central Power and Light Company, dated November 16, 1973.

3. Letter dated December 14, 1973, from D. J. Skovholt to Philip D. Raymond, Niagara Mohawk Power Corporation, with two enclosures: Supplement 1 to Staff Report and Safety Evaluation.

4. Change No. 10 for Nine Mile Point Unit 1, Docket No. 50-220, License No. DPR-17. Letter from D. J. Skovholt to Philip D. Raymond, dated December 28, 1973.

UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-220

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION UNIT 1

NOTICE OF ISSUANCE OF CHANGES TO
TECHNICAL SPECIFICATIONS OF PROVISIONAL OPERATING LICENSE

The Commission issued on February 11, 1974, and published in the Federal Register on February 13, 1974 (39 F.R. 5528), a notice of consideration of a proposed change in the technical specifications of Provisional Operating License No. DPR-17 issued to the Niagara Mohawk Power Corporation to permit the use of fuel assemblies using a partial loading of 8 x 8 fuel (containing U-235) and to authorize changes in the limiting conditions for operations associated with fuel densification of that fuel.

The State of New York, through the Atomic Energy Council of the State of New York, filed a request for leave to intervene as an interested State under 10 CFR 2.715(c) of the Commission's Rules of Practice. No application for leave to intervene, however, has been filed by anyone pursuant to 10 CFR 2.714. On April 8, 1974, the Atomic Safety and Licensing Board, designated to rule on the State's petition to intervene, ordered that the request by the State of New York is deemed to have been withdrawn and the proceeding designated in the Commission's notice of February 11, 1974, is dismissed. Consequently, the Atomic Energy Commission (the Commission) has issued Change No. 11 to the Technical Specifications of Provisional Operating License No. DPR-17 to the Niagara Mohawk Power Corporation (the licensee). This change, effective immediately, authorizes the licensee to operate the Nine Mile Point Nuclear Station Unit 1 (the facility) using 8 x 8 fuel (containing uranium 235) and changes the limiting conditions for operation associated with fuel densification for the 8 x 8 fuel. The licensee is presently authorized to possess and operate its facility located in Oswego County, New York, at power levels up to 1850 MWt using a full core of 7 x 7 fuel (containing uranium 235).

The Commission has found that the application for the above action dated September 14, 1973, as supplemented by filings dated October 15, 1973, January 15, and 22, 1974, and February 19, 1974, complies with the requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission regulations published in 10 CFR Chapter I. The Commission's Directorate of Licensing has completed its evaluation of the action and issued a Safety Evaluation concluding that there is reasonable assurance that the health and safety of the public will not be endangered by the operation of the facility with the 8 x 8 fuel and the related changes to the Technical Specifications as authorized by Change No. 11.

Copies of (1) the Atomic Safety and Licensing Board's Memorandum and Order dated April 8, 1974, (2) Change No. 11 to the Technical Specifications of Provisional Operating License No. DPR-17, (3) the Directorate of Licensing's concurrently issued Safety Evaluation, (4) the Technical Report on the General Electric Company 8 x 8 assembly by the Directorate of Licensing dated February 5, 1974, and (5) the Report of the Advisory Committee on Reactor Safeguards dated February 12, 1974, on the subject of operation of boiling water reactors with 8 x 8 fuel bundles are available for public inspection at the Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C., and at the Oswego City Library at 120 East Second Street, Oswego, New York 13126. Single copies of these items may be obtained upon request sent to the Deputy Director for Reactor Projects, Directorate of Licensing, U. S. Atomic Energy Commission, Washington, D. C. 20545.

FOR THE ATOMIC ENERGY COMMISSION

Dennis L. Ziemann

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

Dated at Bethesda, Maryland,
this 10th day of April 1974.

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Public Proceedings

UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

In the Matter of)
)
NIAGARA MOHAWK POWER CORPORATION) Docket No. 50-220
)
(Nine Mile Point, Unit No. 1))

MEMORANDUM AND ORDER

This Board was established to rule on petitions or requests for leave to intervene in the pending proceeding, which concerns a proposed amendment of the facility operating license previously issued to Niagara Mohawk Power Corporation for the operating of the facility known as Nine Mile Point, Unit 1 (Docket No. 50-220, Provisional Operating License No. DPR-17). The Commission issued on February 11, 1974, and published in the Federal Register on February 13, 1974, a notice of consideration of a proposed change in the technical specifications of the facility operating license to permit the use of fuel assemblies using a partial loading of 8 x 8 fuel (containing U-235) and to authorize changes in the limiting conditions for operations associated with fuel densification of that fuel (39 F.R. 5528).

The State of New York, through the Atomic Energy Council of the State of New York, filed a request for leave to intervene as an interested State under 10 CFR 2.715(c) of the Commission's Rules of Practice. No application for leave to intervene, however, has been filed by anyone pursuant to 10 CFR 2.714.

The Atomic Energy Council of the State of New York has since confirmed, by a letter dated April 1, 1974, that it had wished to have the opportunity to participate in the proceeding only if the Commission should order a hearing on the proposed amendment of the license, either on its own motion or if a licensing board should permit intervention under 10 CFR 2.714. The Council has stated that it did not intend to request a hearing in any other event.

Since the Commission has not ordered a hearing on the amendment on its own motion, and there has been no petition for intervention under 10 CFR 2.714, the application of the Council under Section 2.715(c) is under the circumstances deemed to have been withdrawn.

It is therefore ORDERED that the application of the State of New York for intervention under 10 CFR 2.715(c) is deemed to have been withdrawn, and the proceeding designated in the Commission's notice of hearing dated February 11, 1974 is dismissed.

THE ATOMIC SAFETY AND LICENSING BOARD

Sidney G. Kingsley, Member

Frederick J. Shon, Member

Daniel M. Head, Chairman

Issued at Bethesda, Maryland
this 8th day of April, 1974.