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DEisenhut
ACRS (16)
OPA (CMiles)
DRoss
TBAbernathy
JRBuchanan

Docket No. 50-220

MAY 1 5 1978

Niagara Mohawk Power Corporation
ATTN: Mr. Donald P. Dise
Vice President - Engineering
300 Erie Boulevard West
Syracuse, New York 13202

Gentlemen:

The Commission has issued the enclosed Amendment No. 24 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 April 14, 1977, as supplemented by your letter of April 17, 1978. 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 to permit operation with one recirculating loop isolated.

Copies of the 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. 24 to DPR-63
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

*SEE PREVIOUS YELLOW FOR CONCURRENCES

JJ8

OFFICE	ORB #3	ORB #3	OELD	ORB #3		
SURNAME	*SSheppard:mjf	RClark*	BAB...	GLear G		
DATE	4/26/78	4/27/78	5/10/78	5/15/78		

NRC FORM 318 (9-76) NRCM 0240

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Docket No. 50-220

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ATTN: Mr. Donald P. Dise
Vice President - Engineering
300 Erie Boulevard West
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Sincerely,

George Lear, Chief
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see next page

OFFICE➤	ORB#3	ORB#3	OELD	ORB#3		
SURNAME➤	SSheppard	RClark/ar		GLear		
DATE➤	4/ 26 /78	4/ 27 /78	4/ /78	4/ /78		

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

cc: Eugene B. Thomas, Jr., Esquire
LeBoeuf, Lamb, Leiby & MacRae
1757 N Street, N. W.
Washington, D. C. 20036

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N. W.
Washington, D. C. 20005

T. K. DeBoer, Director
Technological Development Programs
State of New York
Energy Office
Swan Street Building
CORE 1 - Second Floor
Empire State Plaza
Albany, New York 12223

Mr. Robert P. Jones, Supervisor
Town of Scriba
R. D. #4
Oswego, New York 13126

Niagara Mohawk Power Corporation
ATTN: Mr. Thomas Perkins
Plant Superintendent
Nine Mile Point Plant
300 Erie Boulevard West
Syracuse, New York 13202

Chief, Energy Systems Analysis Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, N. W.
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region II Office
ATTN: EIS COORDINATOR
26 Federal Plaza
New York, New York 10007

Oswego County Office Building
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. 15
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 April 14, 1978, 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. 24, 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: May 15, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 24

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.

Remove	Replace
63	63
64a	64a
70	70
70a	70a
70c	70c

3.1.7 FUEL RODS

Applicability:

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective:

The objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specification:

a. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figures 3.1.7.a, 3.1.7.b, 3.1.7.c, 3.1.7.d and 3.1.7.e. If at any time during power operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR 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.

During power operation with one recirculation line isolated, the APLHGR for each fuel type as a function of average planar exposure shall not exceed 98% of limiting value shown in Figures 3.1.7.a, 3.1.7.b, 3.1.7.c, 3.1.7.d and 3.1.7.e.

Amendment No. ~~5~~, ~~16~~, 24

4.1.7 FUEL RODS

Applicability:

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective:

The objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specification:

a. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.

c. Minimum Critical Power Ratio (MCPR)

During power operation MCPR shall be ≥ 1.37 for 7x7 fuel and ≥ 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.

When operating the reactor with one recirculation loop isolated, core power shall be restricted to 90.5% full licensed power.

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.

Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10CFR50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20 F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10CFR50, Appendix K limit. The limiting value for APLHGR is shown in Figure 3.1.7. These curves are based on calculations using the models described in References 1, 2, 3, 5 & 6.

Analysis has been performed (Reference 7) which shows for isolation of 1 loop, operation limited to 98% of the limiting APLHGR shown in Figure 3.1.7 conservatively assures compliance with 10CFR50, Appendix K.

Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation even if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 1 and in References 2, 3 and 4, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at ≥ 25% power to determine if fuel burnup or control rod movement has caused changes in power distribution.

Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25%, the reactor will be operating at a minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal-hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial startup testing 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.

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

(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 USAFC by letter, G. L. Gyorey to Victor Stello Jr., dated December 20, 1974.

(7) September 26, 1975 letter, G. K. Rhode, Niagara Mohawk Power Corporation to G. Lear, United States Nuclear Regulatory Commission.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 24 TO FACILITY OPERATING LICENSE NO. DPR-63

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT UNIT NO. 1

DOCKET NO. 50-220

Introduction

By letter dated April 14, 1977, supplemented by letter dated April 17, 1978, Niagara Mohawk Power Corporation (the licensee) proposed Technical Specification changes for Nine Mile Point Unit No. 1. The changes would permit plant operation with one recirculating loop isolated. The proposed changes were supported by accompanying analyses and by reference to previous analyses provided by the licensee with letters dated September 26, 1975 and October 31, 1975.

Discussion

The Nine Mile Point Nuclear Station Unit No. 1 (NMP-1) contains five (5) reactor coolant recirculation loops, each of which contains a high-capacity motor-driven recirculation pump and two motor-operated gate valves for pump isolation and maintenance. The gate valves are located on the suction and discharge sides of the pumps. Each of the valves on the discharge side of the pumps has a 2 inch bypass around the valve; this bypass is provided with a motor-operated gate valve. The purpose of the bypass is to equalize pressure across the main isolation valves during pump startup and to provide a means of bringing the temperature of the water in the isolated loop up to the bulk reactor water temperature before the main isolation valves are opened.

The NMP-1 reactor was designed to normally have all five recirculation loops in operation. NMP-1 is approved to operate with only four loops in service, with the idle loop not isolated and recirculation flow adjusted to be equivalent to five loop flow to the power-flow corve. If a loop is isolated, this would remove the mass of water in the isolated loop between the two isolated valves from the total vessel coolant inventory. This could affect the results of thermal-hydraulic analyses performed for postulated transient and accident conditions. NMP-1 had not previously performed the necessary analyses with a recirculation loop isolated; hence, the facility is presently authorized to operate with one of the five loops idle but not isolated. The submittal, which is the subject of this safety evaluation, is to justify isolation of an idle loop.

Evaluation

I. Loss of Coolant Accident (LOCA)

A. Idle Loop Not Isolated

For one or more loops out-of-service but not isolated, when the core flow and fluid inventory distribution in the core are the same as would be present with all loops in service, there would be no effect on the ECCS calculation due to the out-of-service loop(s). This is due to the fact that Nine Mile Point does not take credit for extended nucleate boiling caused by flow coast-down in the unbroken loops. Therefore, operation at the full MAPLHGR limits is acceptable so long as the above stated conditions exist.

B. Idle Loop Isolated

The licensee provided the results of calculations with one out-of-service loop's inventory subtracted from the water available for blowdown. The two most limiting small breaks (0.07 and 0.5 ft^2) and the DBA were recalculated with the reduced inventory.

Results of the recalculation revealed that the worst break (highest PCT) didn't change, but for this worst break calculation, it was found that the hot node in the fuel bundle model uncovered a few seconds earlier than with the full inventory available. This earlier uncovering resulted in a slightly higher (30°F) PCT, corresponding to a 1.5% MAPLHGR decrease. The licensee proposed to reduce the MAPLHGR limit by 2% while operating with the idle loop isolated. We find the 2% MAPLHGR limit reduction for one loop out of service and isolated to be acceptable. Calculations were not provided for more than one loop out of service; consequently, such operation is not approved by this SER.

II. One Pump Seizure Accident

The licensee has qualitatively compared the consequences of a pump seizure accident during partial loop operation with a LOCA. Previous analyses have demonstrated that the pump seizure accident is not as severe as a LOCA for fire pump operation. The same conclusion can be made for the four pump cases by analyzing the two events. In both events the recirculating driving loop flow is lost instantaneously: in the case of pump seizure because of pump stoppage and in the LOCA because of a line severance. In the seizure event, natural circulation flow continues, water level is maintained, and the core remains submerged; thus, a continuous core cooling mechanism is provided. However, for a LOCA, complete flow stoppage occurs and the water level decreases resulting in core uncovery and subsequent fuel rod cladding overheating. In addition, the reactor pressure does not decrease for a pump seizure event, whereas complete depressurization occurs for the LOCA. Since the potential effects of a pump seizure accident are bound by the effects of a LOCA, the specific analyses of a pump seizure were not presented by the licensee.

The staff finds the bounding approach for the pump seizure event acceptable.

III. Abnormal Transients

The previous core wide transient analyses for full-loop operation are bounding for partial-loop operation, except for the idle loop startup transient analysis. In the Nine Mile Point-1 FSAR the idle loop startup transient was analyzed with an initial power of 90.5%. Therefore we require that Nine Mile Point-1 restrict its partial-loop operation to a maximum of 90.5% power. Before exceeding 90.5% power during partial-loop operation, Nine Mile Point-1 must submit a revised idle loop startup analysis which is acceptable to the staff.

A large inadvertent flow increase could cause the MCPR to decrease below the Safety Limit MCPR for a low initial MCPR at reduced flow conditions. Therefore, the required MCPR must be increased at reduced core flow by a flow factor, K_f. The K_f-factors are derived assuming all recirculation loops increase speed to the maximum permitted by the scoop tube position set screws. This condition maximizes the power increase and hence the ΔMCPR for transients initiated from less than rated conditions. When operating on four loops, the flow and power increase will be less than with five pumps increasing speed, therefore the K_f factors derived from the five pump assumption are conservative for four loop operation.

The local event rod withdrawal error at rated power analysis indicated that the RBM will stop rod withdrawal at a CPR which is higher than the safety limit. The MCPR requirement for four loop operation will be equal to that for five loop operation because the nuclear characteristics are independent of whether core flow is attained by four or five pump operation.

We find that four loop transients (except the idle loop startup) are bounded by the five loop operation analysis and are therefore acceptable. The present idle loop startup analysis is acceptable for four loop operation up to 90.5% power.

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.

Conclusions

We find that four loop operation up to 90.5% power is acceptable with the fifth loop idle. If the idle loop is isolated, MAPLHGR limits must be reduced by 2%.

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: May 15, 1978

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. 24 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 to permit operation with one recirculating loop isolated.

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 April 14, 1977, (2) Amendment No. 24 to License No. DPR-63, 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 County Library, 46 E. Bridge Street, Oswego, New York 13126. A copy of items (2) and (3) 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 15th day of May 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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