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JUL 1 5 1977

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. 18 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 14, 1977.

The amendment will modify the Technical Specifications to permit operation of the facility on a temporary basis with one emergency cooling system continuously inoperable.

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

Sincerely,

Original signed by

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

Enclosures:
1. Amendment No. 18 to License DPR-63
2. Safety Evaluation
3. Federal Register Notice

cc w/encls:
See next page

Contract

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DATE	7/ 15 /77	7/ 15 /77	7/ 15 /77	7/ 15 /77		

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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. 18
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 14, 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. 18, 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: July 15, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 18

TO THE TECHNICAL SPECIFICATIONS

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 page is identified by Amendment number and contains vertical lines indicating the area of change. Add page 47a.

Remove	Replace
47	47 47a
48	48

3.1.3 EMERGENCY COOLING SYSTEM

Applicability:

Applies to the operating status of the emergency cooling system.

Objective:

To assure the capability of the emergency cooling system to cool the reactor coolant in the event the normal reactor heat sink is not available.

Specification:

a. During power operating conditions and whenever the reactor coolant temperature is greater than 212F, both emergency cooling systems shall be operable except as specified in 3.1.3.b and c.

b. **During Cycle 5, with one emergency** cooling system inoperable, specification 3.1.3.a shall be considered fulfilled, provided that the inoperable system is returned to an operable condition at the first cold shutdown after August 11, 1977 and the additional surveillance required is performed. Subsequent operation including the remainder of Cycle 5, shall comply with specification 3.1.3.c.

4.1.3 EMERGENCY COOLING SYSTEM

Applicability:

Applies to periodic testing requirements for the emergency cooling system.

Objective:

To assure the capability of the emergency cooling system for cooling of the reactor coolant.

Specification:

The emergency cooling system surveillance shall be performed as indicated below:

a. At least once every five years -

The system heat removal capability shall be determined.

b. At least once daily -

The shell side water level and makeup tank water level shall be checked.

c. At least once per month -

The makeup tank level control valve shall be manually opened and closed.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. During Cycle 6 and subsequent cycles if one emergency cooling system becomes inoperable, Specification 3.1.3.a shall be considered fulfilled, provided that the inoperable system is returned to an operable condition within 7 days and the additional surveillance required is performed.	

LIMITING CONDITION FOR OPERATION

d. Makeup water shall be available from the two gravity feed makeup water tanks.

e. If Specifications 3.1.3a, b, c or d are not met, a normal orderly shutdown shall be initiated within one hour and the reactor shall be in the cold shutdown condition within ten hours.

SURVEILLANCE REQUIREMENT

d. At least once each shift -

The area temperature shall be checked.

e. During each major refueling outage -

Automatic actuation and functional system testing shall be performed during each major refueling outage and whenever major repairs are completed on the system.

f. Surveillance with an Inoperable System

During Cycle 5 with one of the emergency cooling systems inoperable and specification 3.1.3.b. in affect, the level control valve and the motor-operated isolation valve in the operable system shall be demonstrated to be operable immediately and weekly thereafter.

During Cycle 6 and subsequent cycles, when one of the emergency cooling systems is inoperable, the level control valve and the motor-operated isolation valve in the operable system shall be demonstrated to be operable immediately and daily thereafter.

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

SAFETY EVALUTION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 18 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 July 14, 1977, Niagara Mohawk Power Corporation (NMPC) requested an amendment to Facility Operating License No. DPR-63. The amendment would modify the Technical Specifications for the Nine Mile Point Unit No. 1 (NMP-1) to permit operation of the facility on a temporary basis with one "emergency cooling system", i.e., isolation condenser, continuously inoperable.

Background

The licensee had begun Cycle 5 operation of NMP-1 on July 10, 1977, at which time examination of the two emergency cooling systems (isolation condensers) revealed a broken valve stem in the return line of one of the emergency cooling systems. The broken valve stem renders one of the emergency cooling systems inoperable. The current Technical Specifications requires that the inoperable system be returned to an operable condition within 7 days or a normal orderly shutdown shall be initiated.

The licensee has stated that the necessary parts for the valve repair are not readily available and after discussion with their suppliers they have determined that the earliest delivery of the required parts would be two to three weeks. Because of the inaccessibility of the return line and the configuration of the system, the repair is expected to take up to 2 weeks once the required parts are made available. Because of the long repair time involved, NMP-1 has proposed to operate with one emergency cooling system (isolation condenser) inoperable for the duration of Cycle 5.

Evaluation

The emergency cooling system (isolation condensers) use a network of piping through which primary steam can be circulated. The network of piping in each system is submerged in cooling water in a large tank initially containing over 20,000 gallons of water with over 70,000 additional gallons available from the condensate storage tanks. The system condenses the primary steam within piping in the tank and returns the condensate by gravity flow to one of the recirculating water loops. The isolation condenser is therefore a passive system except for valves which must be operated to open the submerged piping network to the steam flow. These two systems remove primary system decay heat when the primary system is at saturated conditions and is isolated from the main condenser. Each of the two systems is capable of removing about 3% of full core power.

We have considered the proposed changes as they affect all significant areas: (1) effect on Emergency Core Cooling Systems (ECCS) following a postulated loss-of-coolant accident (LOCA), (2) effect on transient analyses, and (3) effect on reactor shutdown, both normal and with the primary system isolated from the main condenser. Bases for acceptability are given in the following sections:

Emergency Core Cooling with One Inoperable Emergency Cooling System

The Emergency Core Cooling System postulated LOCA analyses for the limiting break location[1] (recirculation line) and for the non-limiting steam line break submitted by NMP-1 for the current cycle assume: (a) single failure of one isolation condenser system; (b) loss of the other isolation condenser system due to break location (this occurs for a break location at either (1) the junction of the steam line and the isolation condenser system steam line or (2) at the junction of the recirculation line and isolation condenser system condensate return line); and (c) an additional conservative assumption that the worst single failure has occurred in the automatic-depressurization-system (ADS) which delays ADS actuation by 5 seconds. The assumption of the two single failures mentioned above, i.e., failure of one isolation condenser plus the ADS 5 second delay, was beyond the requirements for such ECCS analyses (worst single failure). However, the analyses were performed this way for

1/ Limiting Break Location may be defined as the location of a break for a LOCA that results in the highest peak clad temperature (PCT).

calculational ease: each of the failures has only a very small effect on the allowable operating power, and by assuming both failures it was not necessary to determine, for each break size, which of the two failures was most limiting. Fortuitously, however, those analyses are acceptable for the present situation with one emergency condenser inoperable. That is, it is now required that one isolation condenser be unavailable, that the other isolation condenser be lost due to break location, and that the worst remaining single failure (ADS 5 second delay) be assumed. These are exactly the conditions that have already been analysed. Therefore, the additional system's unavailability has already been considered (previously as an additional unrequired "single failure", but nevertheless considered) and the limiting break analyses and the MAPLHGR limits are therefore unchanged by unavailability of one isolation condenser.

For other (non-limiting location) line breaks (feedwater and core spray) the previously approved analyses assumed failure of one isolation condenser system plus the worst ADS failure (resulting in a 5 second delay). Therefore, in those previous analyses, credit was taken for availability of one isolation condenser system (the break location in those cases cannot directly disable an isolation condenser system). In the present case, with one isolation condenser system inoperable, the worst single failure is either failure of the other isolation condenser system or the ADS 5 second delay discussed above. The previous analyses cover only the case with the ADS delay with one isolation condenser still available. It is possible that the other case, failure of the second isolation condenser, could result in slightly higher clad temperatures; however, previous sensitivity studies have shown that for feedwater or core spray breaks, the peak clad temperature (PCT) will vary less than 50°F due to loss of an isolation condenser system. Therefore, since these breaks are not at the limiting break locations (their PCT are more than 300° below the PCT for the limiting break), it is concluded that the assumption of both isolation condensers being unavailable cannot cause these analyses to become limiting.

Since the previous limiting break analyses already considered the present cycle with both isolation condenser systems inoperable plus the worst single failure, and since the assumption of both isolation condensers being inoperable in the non-limiting break locations analyses cannot cause those locations to become limiting, we conclude that the LOCA analyses submitted by the licensee for the current cycle are acceptable for operation with one isolation condenser system inoperable.

Abnormal Operational Transients

It has been determined that previous transient results remain applicable with one emergency cooling system, i.e., isolation condenser, continuously inoperable, since credit was taken for only one emergency cooling system in these analyses.

Shutdown

For normal shutdown, the main condenser is available for an extended period and is used to remove decay heat until the shutdown cooling system is utilized. With the exception of only one or two shutdowns during the plant life to date (startup was in 1969), all shutdowns have had the main condenser available so the isolation condenser systems were not needed and were not utilized. Therefore, operation with one isolation condenser unavailable is acceptable for normal shutdown.

For shutdown when the reactor primary system is isolated (i.e., the main condenser is not available), the isolation condenser systems are used to remove decay heat. Each isolation condenser system is capable of removing about 3% of full reactor core power. Until decay heat is reduced to a level where the isolation condenser system(s) are capable of removing the decay heat, the heat must be expelled as steam through the relief valves. Therefore, with both isolation condenser systems available, 6% of core power can be removed through the isolation condenser systems. This level of decay heat is reached in 10 to 20 seconds following a scram. However, with only one isolation condenser system available, 3% of core power can be removed through the one isolation condenser system. This condition is not reached until between 200 and 400 seconds following a scram. Therefore, the effect of unavailability of one isolation condenser system is several minutes of additional time during which steam must be discharged intermittently to the torus through the relief valves. Considering the low probability of an isolation shutdown and the fact that the availability of both isolation condenser systems does not prevent such discharges but merely decreases their duration, it is concluded that the unavailability of one isolation condenser system contributes only a small increase in the expected plant lifetime relief valve discharge into the torus. Such discharges of water into the torus are anticipated occasionally during plant operation and present no significant safety hazard.

We therefore conclude that plant shutdown both with and without isolation of the primary system is acceptable with an isolation condenser system out of service.

Conclusions

We conclude that operation for an extended period is acceptable with one isolation condenser system inoperable, on the bases stated above. We further conclude that the surveillance requirements on the remaining operable isolation condenser system should be changed from daily (currently required if an isolation condenser is inoperable) to weekly, as proposed by NMP-1. Daily testing of the operable system's valves for an extended period would not be appropriate due to wear considerations.

However, we require that NMP-1 restore the inoperable isolation condenser system to operable status at the first cold shutdown after August 11, 1977, if such conditions exist before the next refueling outage (at which time **NMP-1 had proposed such restoration). This change has been made to the Technical Specifications after discussion and agreement with** NMPC.

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 environmnetal impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusions

We have concluded, based on the considerations discussed above, 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, (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: **July 15, 1977**

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-220

NIAGARA MOHAWK POWER CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 18 to Facility Operating License No. DPR-63 for 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 consists of changes to the Technical Specifications which will permit operation of the facility on a temporary basis with one emergency cooling system (isolation condenser) continuously inoperable.

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 or negative declaration and 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 14, 1977, (2) Amendment No. 18 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 City 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 15 day of July 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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