AUG 15 1975

Docket No. 50-220

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Niagara Mohawk Power Corporation
ATTN: Mr. Gerald K. Rhode
Vice President - Engineering
300 Erie Boulevard West
Syracuse, New York 13202

Gentlemen:

The Commission has requested the Federal Register to publish the enclosed Notice of Proposed Issuance of an Amendment to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station Unit 1. The proposed amendment includes a change to the Technical Specifications and reflects your acceptance, by letter dated July 2, 1975, of our proposal of June 13, 1975.

This amendment incorporates: (1) water temperature limits during any testing which adds heat to the suppression pool, (2) suppression pool water temperature limits requiring manual scram of the reactor, (3) suppression pool water temperature limits requring reactor pressure vessel depressurization, (4) surveillance requirements to monitor water temperatures during operations which add heat to the suppression pool and (5) external visual examinations of the suppression chambers following operations in which the pool temperatures exceed 160°F.

Copies of the proposed amendment, the related Safety Evaluation, and the Federal Register Notice are enclosed.

Sincerely,

/S/

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

Enclosures:
1. Federal Register Notice
2. Proposed Amendment
3. Safety Evaluation

ccs: See next page

OFFICE	ORB#3	ORB#3	OELD	ORB#3		
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DATE	8/ 1 /75	8/ 1 /75	8/ 12 /75	8/ 14 /75		

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Vice President - Engineering
300 Erie Boulevard West
Syracuse, New York 13202

Gentlemen:

The Commission has requested the Federal Register to publish the enclosed Notice of Proposed Issuance of an Amendment to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station Unit 1.

The proposed amendment would define new temperature limits for the suppression pool water to provide additional assurance of maintaining primary containment integrity. This action reflects your acceptance, by letter dated July 2, 1975, of our proposal of June 13, 1975.

A copy of the proposed Amendment and the related Safety Evaluation is enclosed.

Sincerely,

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

Enclosures:
1. Federal Register Notice
2. Proposed Amendment
3. Safety Evaluation

cc w/encls:
See next page

W. PAULSON
RL:ORB-3
WAP 8/4/75

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DATE ➤	8/1/75	8/1/75	8/ /75	8/ /75		

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

Niagara Mohawk Power Corporation

AUG 1 5 1975

cc: w/enclosures

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LeBoeuf, Lamb Leiby & MacRae
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Anthony Z. Roisman, Esquire
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New York State Atomic Energy Council
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Docket No. 50-220

AUG 1 5 1975

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ATTN: Mr. Gerald K. Rhode
Vice President - Engineering
300 Erie Boulevard West
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Gentlemen:

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This amendment incorporates: (1) water temperature limits during any testing which adds heat to the suppression pool, (2) suppression pool water temperature limits requiring manual scram of the reactor, (3) suppression pool water temperature limits requring reactor pressure vessel depressurization, (4) surveillance requirements to monitor water temperatures during operations which add heat to the suppression pool and (5) external visual examinations of the suppression chambers following operations in which the pool temperatures exceed 160°F.

Copies of the proposed amendment, the related Safety Evaluation, and the Federal Register Notice are enclosed.

Sincerely,

George Lear

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

Enclosures:
1. Federal Register Notice
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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 1

PROPOSED AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.
License No. DPR-63

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. 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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility 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, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. ".

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 Reactor Licensing

Attachment:
Change No. to the
Technical Specifications

Date of Issuance:

ATTACHMENT TO PROPOSED AMENDMENT NO.

CHANGE NO.   TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

Replace pages 129, 130 and 134 with the attached revised pages. Add page 134-a.

SSURE SUPPRESSION SYSTEM PRESSURE AND PRESSION CHAMBER WATER TEMPERATURE LEVEL

licability:

lies to the interrelated parameters of ssure suppression system pressure and pression chamber water temperature l level.

ective:

assure that the peak suppression mber pressure does not exceed design ues in the event of a loss-of-coolant cident.

cification:

The downcomers in the suppression chamber shall have a minimum submergence of three feet and a maximum submergence of five feet whenever the reactor coolant system temperature is above 215F.

During normal power operation, the combination of primary containment pressure and suppression chamber water temperature shall be within the shaded area of (1) Figure 3.3.2a when downcomer submergence is 5 feet, (2)

4.3.2 PRESSURE SUPPRESSION SYSTEM PRESSURE AND SUPPRESSION CHAMBER WATER TEMPERATURE AND LEVEL

Applicability:

Applies to the periodic testing of the pressure suppression system pressure and suppression chamber water temperature and level.

Objective:

To assure that the pressure suppression system pressure and suppression chamber water temperature and level are within required limits.

Specification:

a. At least once per day the suppression chamber water level and temperature and pressure suppression system pressure shall be checked.

b. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

c. Whenever heat from relief valve operation is being added to the suppression pool the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

Figure 3.3.2b when downcomer submergence is $\geq$ 4 feet, or (3) Figure 3.3.2c when downcomer submergence is $\geq$ 3 feet. If these temperatures are exceeded, pool cooling shall be initiated immediately.

c. If Specifications a and b above are not met within 24 hours, the reactor shall be shutdown using normal shutdown procedures.

d. During testing of relief valves which add heat to the torus pool, the water temperature shall not exceed 10F above the normal power operation limit specified in b above. In connection with such testing the pool temperature must be reduced within 24 hours to below the normal power operation limit specified in b above.

e. The reactor shall be scrammed from any operating condition when the suppression pool temperature reaches 110F. Operation shall not be resumed until the pool temperature is reduced to below the normal power operation limit specified in b above.

f. During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the pool temperature reaches 120F.

d. Whenever operation of a relief valve is indicated and the suppression pool temperature reaches 160F or above while the reactor primary coolant system pressure is greater than 200 psig, an external visual examination of the suppression chamber shall be made before resuming normal power operation.

'he values specified for suppression chamber water temperature, maximum downcomer submergence, and system ressures are based on the effect these parameters have on the short-term post-accident system pressure 'ollowing a loss-of-coolant accident. The combinations shown on Figures 3.3.2 a, b and c and the water level equired are based on maintaining the post-accident pressure below the design value of 35 psig and the iaximum suppression chamber water temperature below 140F in the containment design basis loss-of-coolant ccident (Appendix E-11.2.2.3).*

'he calculational basis for the pressure suppression system initial conditions, Figures 3.3.2 a, b and c are resented in the Fifth Supplement.*

'he three foot minimum and the five foot maximum submergence are a result of the Moss Landing Tests reported n Volume I of the PHSR under "Pressure Suppression Design Basis".

'he 215F limit for the reactor is specified, since below this temperature the containment can tolerate a lowdown without exceeding the 35 psig design pressure of the suppression chamber without condensation.

ctually, for reactor temperatures up to 312F the containment can tolerate a blowdown without exceeding the 5 psig design pressure of the suppression chamber, without condensation.

xperimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of he suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions t the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that he reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber oadings.

n addition to the limits on temperature of the suppression chamber pool water, operating procedures define he action to be taken in the event a relief valve inadvertently opens or sticks open. This ction would include: (1) use of all available means to close the valve, (2) initiate suppression pool water ooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to lepressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to issure mixing and uniformity of energy insertion to the pool.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings

*FSAR

could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

Continuous monitoring of suppression chamber water level and temperature and pressure suppression system pressure is provided in the control room. Alarms for these parameters are also provided in the control room.

To determine the status of the pressure suppression system, inspections of the suppression chamber interior surfaces at each major refueling outage with water at its normal elevation will be made. This will assure that gross defects are not developing.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT TO LICENSE NO. DPR-63
AND CHANGE TO TECHNICAL SPECIFICATIONS
SUPPRESSION POOL WATER TEMPERATURE LIMITS
NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT UNIT 1
DOCKET NO. 50-220

Introduction

By letter dated February 15, 1975 to Niagara Mohawk Power Corporation, the Nuclear Regulatory Commission (NRC) requested that the licensee among other things, develop operating procedures and proposed changes to the Technical Specifications to preclude reaching elevated temperatures of the torus pool water and to provide for inspection of the torus as appropriate to identify any damage in the event of an extended relief valve operation. By letter dated March 27, 1975 Niagara Mohawk submitted a response which stated that the present Technical Specifications provide adequate limits for the suppression chamber water temperature, thus the licensee proposed no change to the Technical Specifications. This response was found to be unacceptable; and, as a result, the NRC staff prepared appropriate technical specification changes to revise the suppression pool water temperature limits for Nine Mile Point Unit 1. By letter dated June 13, 1975, the NRC staff advised the licensee of its intent to intiate steps to issue these technical specification changes unless the licensee objected in writing. By letter dated July 2, 1975, the licensee replied that they had no objection to the incorporation of the proposed technical specification changes into the existing Technical Specifications for Nine Mile Point Unit 1.

Discussion

Nine Mile Point Unit 1 is a boiling water reactor (BWR) which is housed in a Mark I primary containment. The Mark I primary containment is a pressure suppression type of primary containment that consists of a drywell and a suppression chamber (also referred to as the torus). The suppression chamber, or torus, contains a pool of water and is designed to suppress the pressure during a postulated loss-of-coolant accident (LOCA) by condensing the steam released from the reactor primary system. The reactor system energy released by relief valve operation during operating transients also is released into the pool of water in the torus.

Experiences at various BWR plants with Mark I Containments have shown that damage to the torus structure can occur from two phenomena associated with relief valve operations. Damage can result from the forces exerted on the structure when, on first opening the relief valves, steam and the air within the vent are discharged into the torus water. This phenomenon is referred to as steam vent clearing. The second source of potential structural damage stems from the vibrations which accompany extended relief valve discharge into the torus water if the pool water is at elevated temperatures. This effect is known as the steam quenching vibration phenomenon.

A. Steam Vent Clearing Phenomenon

With regard to the steam vent clearing phenomenon, we are actively reviewing this generic problem and in our letter dated February 15, 1975 we also requested the licensee to provide information to demonstrate that the torus structure of the primary containment will maintain its integrity throughout the anticipated life of the facility. In its response dated March 27, 1975 the licensee stated that it was investigating this matter and the results of the investigation would be submitted to us on a schedule consistent with the timing which we proposed for licensee response. Because of the apparent slow progression of the material fatigue associated with the steam vent clearing phenomenon, we have concluded that there is no immediate potential hazard resulting from this type of phenomenon; nevertheless, surveillance and review action on this matter by the NRC staff will continue in due course during this year.

B. Steam Quenching Vibration Phenomenon

The steam quenching vibration phenomenon became a concern as a result of occurrences at two European reactors. With torus pool water temperatures increased in excess of 170F due to prolonged steam quenching from relief valve operation, hydrodynamic fluid vibrations occurred with subsequent moderate to high relief valve flow rates. These fluid vibrations produced large dynamic loads on the torus structure and extensive damage to torus internal structures. If allowed to continue, the dynamic loads could have resulted in structural damage to the torus itself, due to material fatigue. Thus, the reported occurrences of the steam quenching vibration phenomenon at the two European reactors indicate that actual or incipient failure of the torus can occur from such an event. Such failure would be expected to involve cracking of the torus wall and loss of containment integrity. Moreover, if a LOCA occurred simultaneously with or after such an event, the consequences could be excessive radiological doses to the public. In comparison with

the steam vent clearing phenomenon, the potential risk associated with the steam quenching vibration pheonomenon (1) reflects the fact that a generally smaller safety margin[1/] exists between the present license requirements on suppression pool temperature limits and the point at which damage could begin and (2) is more immediate.

Evaluation

The existing Technical Specifications for the torus pool water establish temperature limits that are functions of downcomer submergence (i.e., pool depth) and containment system pressure. These limits are presented in Figures 3.3.2a, b, and c of the Technical Specifications for a selected number of downcomer submergences based upon the capability of the pool water to maintain post-accident pressure below the containment design pressure and the maximum suppression chamber water temperature below 140F during the loss-of-coolant accident. The upper limit on water temperature permitted by this family of curves is 94F for a 5 ft submergence. While this family of curves provides normal operating flexibility, a short-term temperature limit of 130F permitted by operating procedures exceeds the normal power operating temperature limit, but accommodates the heat release resulting from abnormal operation, such as relief valve malfunction, while still maintaining the required heat-sink (absorption) capacity of the pool water needed for the postulated LOCA conditions. However, in view of the potential risk associated with the steam quenching vibration phenomenon, it is necessary to modify the temperature limits now in the license Technical Specifications. This action was, as discussed in our February 15, 1975 letter, first suggested by the General Electric Company (GE) who had earlier informed us of the steam quenching vibration occurrences at a meeting on November 1, 1974 and provided related information by letters to us dated November 7, and December 20, 1974. The December 20 letter stated that GE had informed all of its customers with operating BWR facilities and Mark I containments of the phenomenon and included in those communications GE's recommended interim operating temperature limits and proposed operating procedures to minimize the probability of encountering the damaging regime of the steam quenching vibration phenomenon.

Our implementation of the GE recommended procedures and temperature limits via changes in the Technical Specifications are evaluated in the following paragraphs:

1/ The difference, in pool water temperature, between the license limit(s) and the temperature at which structural damage might occur is the safety margin available to protect against the effects of the phenomenon discussed.

a. The new short-term limit applicable to all conditions requires that the reactor be scrammed if the torus pool water temperature reaches 110F. This new limit and associated requirement to scram the reactor provides additional margin below the 170F temperature related to potential damage to the torus. Since the current operating procedures permit the torus pool water temperature to reach 130F in the event of a relief valve malfunction before requiring the reactor to be scrammed, reducing this limit to 110F provides an additional margin of 20F for absorption of reactor core decay heat.

b. For specific requirements associated with surveillance testing, i.e., testing of relief valves, the water temperature shall not exceed 10F above the normal power operation limit. This new limit during surveillance testing of relief valves provides additional operating flexibility while still maintaining a maximum heat-sink capacity. The current limits in the Technical Specifications make no provision for these requirements.

c. For reactor isolation conditions, the new temperature limit is 120F, above which temperature the reactor vessel is to be depressurized. This new limit of 120F assures pool capacity for absorption of heat released to the torus while avoiding undesirable reactor vessel cooldown transients. Upon reaching 120F, the reactor is placed in the cold, shutdown condition at the fastest rate consistent with the technical specifications on reactor pressure vessel cooldown rates.

d. In addition to the new limits on temperature of the torus pool water, the discussion in the Basis includes a summary of required operator actions to be taken in the event of a relief valve malfunction. These operating actions are taken in order to avoid the development of temperatures approaching the 170F threshhold for potential damage by the steam quenching phenomenon.

Conclusion

We have concluded, based on the consideration discussed above that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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: AUG 1 5 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-220

NIAGARA MOHAWK POWER CORPORATION

NOTICE OF PROPOSED ISSUANCE OF AMENDMENT

TO FACILITY OPERATING LICENSE

The Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. DPR-63 issued to Niagara Mohawk Power Corporation (the licensee), for operation of the Nine Mile Point Nuclear Station, Unit 1, located in Oswego County, New York.

The amendment would revise the provisions in the Technical Specifications relating to temperature limits for the pressure suppression pool water.

Prior to issuance of the proposed license amendment, the Commission will have made the findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations.

By September 24, 1975 the licensee may file a request for a hearing and any person whose interest may be affected by this proceeding may file a request for a hearing in the form of a petition for leave to intervene with respect to the issuance of the amendment to the subject facility operating license. Petitions for leave to intervene must be filed under oath or affirmation in accordance with the provisions of Section 2.714 of 10 CFR Part 2 of the Commission's regulations. A petition for leave to intervene must set forth the interest of the

petitioner in the proceeding, how that interest may be affected by the results of the proceeding, and the petitioner's contentions with respect to the proposed licensing action. Such petitions must be filed in accordance with the provisions of this FEDERAL REGISTER notice and Section 2.714, and must be filed with the Secretary of the Commission, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Docketing and Service Section, by the above date. A copy of the petition and/or request for a hearing should be sent to the Executive Legal Director, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, and to Arvin E. Upton, Esq., LeBoeuf, Lamb, Leiby & MacRae, 1757 N Street, N. W., Washington, D. C. 20036, the attorney for the licensee.

A petition for leave to intervene must be accompanied by a supporting affidavit which identifies the specific aspect or aspects of the proceeding as to which intervention is desired and specifies with particularity the facts on which the petitioner relies as to both his interest and his contentions with regard to each aspect on which intervention is requested. Petitions stating contentions relating only to matters outside the Commission's jurisdiction will be denied.

All petitions will be acted upon by the Commission or licensing board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel. Timely petitions will be considered to determine whether a hearing should be noticed or another appropriate order issued regarding the disposition of the petitions.

In the event that a hearing is held and a person is permitted to intervene, he becomes a party to the proceeding and has a right to participate fully in the conduct of the hearing. For example, he may present evidence and examine and cross-examine witnesses.

For further details with respect to this action, see the letter from K. Goller to G. Rhode dated June 13, 1975 and the letter from G. Rhode to K. Goller dated July 2, 1975, which 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, 120 E. Second Street, Oswego, New York 13126. The proposed license amendment and the Safety Evaluation, may be inspected at the above locations and a copy may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 15th day of [illegible], [illegible].

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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