JUL 1 5 1975

Docket No. 50-263

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bcc: JRBuchanan
TBAbernathy

Northern States Power Company
ATTN: Mr. L. O. Mayer
Director of Nuclear
Support Services
414 Nicollet Mall
Minneapolis, Minnesota 55401

Gentlemen:

The Commission has requested the Federal Register to publish the enclosed Notice of Proposed Issuance of an Amendment to Facility License No. DPR-22 for the Monticello Nuclear Generating Plant. The proposed amendment includes a change to the Technical Specifications and is in response to your request dated March 24, 1975, which was submitted in reply to our letter dated February 14, 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 requiring 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.

During our review, we discussed with your staff certain modifications to the proposed change which they agreed were necessary for clarification and completeness. These modifications have been made.

Copies of our proposed license amendment with changes to the Technical Specifications, Safety Evaluation and the Federal Register Notice relating to this action also are enclosed.

Sincerely,

Original Signed by:
Dennis L. Ziemann

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

O/P

Enclosures:
1. Proposed Amendment
w/Proposed Tech Spec change
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:
See next page

OELD S. H. Lewis 7/8/75

OFFICE	RL:ORB#2	RL:ORB#2	RL:ORB#3	RL:ORB#2	RL:AD/ORs
SURNAME	RMDiggs:tc	BCBuckley	CJDeBevec	DLZiemann	KRGoller
DATE	6/25/75	7/1/75	7/2/75	7/2/75	7/15/75

Form AEC-318 (Rev. 9-53) AECM 0240

cc w/enclosures:
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Minneapolis, Minnesota 55401

Mr. D. S. Douglas, Auditor
Wright County Board of Commissioners
Buffalo, Minnesota 55313

cc w/enclosures and cy of NSP's
filing dtd. 3/24/75:
Warren R. Lawson, M. D.
Secretary and Executive Officer
State Department of Health
University Campus
Minneapolis, Minnesota 55440

Mr. Gary Williams
Federal Activities Branch
Environmental Protection Agency
230 South Dearborn Street
Chicago, Illinois 60604

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-271

MONTICELLO NUCLEAR GENERATING STATION

PROPOSED AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.
License No. DPR-22

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

   A. The application for amendment by Northern States Power Company (the licensee) dated March 24, 1975, 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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Facility License No. DPR-22 is hereby amended to read as follows:

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"B. Technical Specifications

The Technical Specifications contained in Appendix A, 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

A. Giambusso, Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Change No. to the
Technical Specifications

Date of Issuance:

OFFICE ➤						
SURNAME ➤						
DATE ➤						

☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

PROPOSED CHANGE TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

The Technical Specifications contained in Appendix A, attached to Facility Operating License No. DPR-22, are hereby changed by replacing pages 139, 140, 157 and 161 with revised pages bearing the same numbers and additional pages 157A and 161A. Changed areas on the revised pages are reflected by marginal lines.

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3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment.

1. At any time that the nuclear system is pressurized above atmospheric or work is being done which has the potential to drain the vessel, except as permitted by specification 3.5.G.4, the suppression pool water volume and temperature shall be maintained within the following limits.

   (a) Maximum Water Temperature during normal operation 90°F.

   (b) Maximum Water Temperature during any test operation which adds heat to the suppression pool - 100°F and shall not be above 90°F for more than 24 hours.

   (c) If Torus Water Temperature exceeds 110°F, initiate an immediate scram of the reactor. Power operation shall not be resumed until the pool temperature is reduced below 90°F.

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment.

1. The suppression chamber water level and temperature shall be checked once per day. A visual inspection of the suppression chamber interior including water line regions and the interior painted surfaces above the water line shall be made at each refueling outage. Whenever there is indication of relief valve operation which adds heat 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. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an extended visual examination of the suppression chamber shall be conducted before resuming power operation.

(d) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the torus water temperature exceeds 120°F.

(e) Minimum Water Volume 68,000 cubic feet.

(f) Maximum Water Volume 77,970 cubic feet.

2. Primary containment integrity as defined in the Section 1, shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 Mw(t).

2. The primary containment integrity shall be demonstrated as follows:

(a) Integrated Primary Containment Leak Test (IPCLT)

(1) An integrated leak rate test shall be performed prior to initial unit operation at an initial test pressure (Pt) of 41 psig.

(2) Subsequent leak rate tests shall be performed without preliminary leak detection surveys or leak repairs immediately prior to or during the test, at an initial pressure of approximately 41 psig.

(3) Leak repairs, if necessary to permit integrated leak rate testing, shall be preceded by local leak rate measurements where possible. The leak rate differ-

Bases Continued:

3.7 A. Primary Containment

length of four feet, which resulted in complete condensation. Thus with respect to downcomer submergence, this specification is adequate.

The maximum temperature at the end of blowdown tested during the Humboldt Bay [1] and Bodega Bay[2] tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

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

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

For an initial maximum suppression chamber water temperature of 90°F and assuming the normal complement of containment cooling pumps (2 LPCI pumps and 2 containment cooling service water pumps). containment pressure is not required to maintain adequate net positive suction head (NPSH) for the core spray, LPCI and HPCI pumps. However, during an approximately one-day period starting a few hours after a loss-of-coolant accident, should one RHR loop be inoperable and should the containment pressure be reduced to atmospheric pressure through any means, adequate NPSH would not be available. Since an extremely degraded condition must exist, the period of vulnerability to this event is restricted by Specification 3.7.A.1.b by limiting the suppression pool initial temperature and the period of operation with one inoperable RHR loop.

(1) Robbins, C. H., "Tests of Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.

(2) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

Bases Continued:

3.7 A. Primary Containment

If a loss of coolant accident were to occur when the reactor water temperature is below 330°F, the containment pressure will not exceed the 62 psig design pressure, even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperatures above 212°F provides additional margin above that available at 330°F.

Bases:

4.7 A. Primary Containment

The water in the suppression chamber is used only for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a weekly check of the temperature and volume is adequate to assure that adequate heat removal capability is present. For additional margin, these will be checked once per day.

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, approximately once per year, assures the paint is intact and is not deteriorating. Experience with this type of paint indicates that the inspection interval is adequate.

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 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. Visual inspection of the suppression chamber including water line regions each refueling outage is adequate to detect any changes in the suppression chamber structures.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss of coolant accident. The peak drywell pressure would be about 41 psig, which would rapidly reduce to 25 psig within 10 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 25 psig within 10 seconds, equalizes with drywell pressure and thereafter rapidly decays with the drywell pressure decay. See Section 5.2.3 FSAR.

The design pressure of the drywell and absorption chamber is 56 psig. See Section 5.2.3 FSAR. The design leak rate is 0.5%/day at a pressure of 56 psig. As indicated above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 10 seconds. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressures were chosen. Also, based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

Bases Continued:

4.7 A. Primary Containment

The design basis loss of coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.5% day at 41 psig. The analysis showed that with this leak

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT TO LICENSE NO. DPR-22
AND
CHANGE TO THE TECHNICAL SPECIFICATIONS

SUPPRESSION POOL WATER TEMPERATURE LIMITS

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

INTRODUCTION

By letter dated March 24, 1975, Northern States Power Company (NSPC) requested a change in the Technical Specifications appended to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant located in Wright County, Minnesota. The proposed change in Technical Specifications was submitted in response to our request to the licensee dated February 14, 1975, and is responsive to the guidelines set forth in our letter. We have made additional modifications to these proposed Technical Specifications to improve the clarity and intent of the specification and its basis. These additional changes were discussed with and agreed to by the NSPC staff members. The proposed change in Technical Specifications defines new temperature limits for the suppression pool water to provide additional assurance of maintaining primary containment function and integrity in the event of extended relief valve operation.

DISCUSSION

The Monticello plant 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.

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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.

1. 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 14, 1975, we also requested each applicable licensee to provide information to demonstrate that the torus structure will maintain its integrity throughout the anticipated life of the facility. Because of apparent slow progression of the material fatigue associated with the steam vent clearing phenomenon, we have concluded that there is not immediate potential hazard resulting from this type of phenomenon; nevertheless, surveillance and review action on this matter by the NRC staff will continue during this year.

2. 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 170° F 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 in 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.

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In comparison with the steam vent clearing phenomenon, the potential risk associated with the steam quenching vibration phenomenon (1) reflects the fact that a generally smaller safety margin1/ 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 Monticello plant limit the torus pool temperature to 90°F. This temperature limit assures that the pool water has the capability to perform as a constantly available heat-sink with a reasonable operating temperature that can be maintained by use of heat exchangers whose secondary cooling water (the service cooling water) is expected to remain well below 90°F. While this 90°F limit provides normal operating flexibility, short-term temperatures permitted by operating procedures exceed 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 in the Technical Specifications.

This action was, as discussed in our February 14, 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 letter of December 20, 1974 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.

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a. The new short-term temperature limit applicable to all reactor operating conditions requires that the reactor be scrammed if the torus pool water temperature exceeds 110°F. This new temperature limit and associated requirement to scram the reactor provides an additional safety margin below the 170°F temperatures related to potential damage to the torus.

b. For specific requirements associated with surveillance testing, i.e., testing of relief valves, HPCI and RCIC, the water temperature shall not exceed 100°F, i.e., 10°F above the normal power operation limit. This new limit applicable to surveillance testing provides additional operating flexibility while still maintaining a maximum heat-sink capacity. The current limits in the Technical Specifications made a provision for these requirements but were less restrictive on the maximum water temperature, i.e., current limit is 130°F. The time allowed for return to normal operating temperature is unchanged.

c. For reactor isolation conditions, the new temperature limit is 120°F, above which temperature the reactor vessel is to be depressurized. This new limit of 120°F assures pool capacity for absorption of heat released to the torus while avoiding undesirable reactor vessel cooldown transients. Upon reaching 120°F, 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, discussion in the Basis includes a summary of operator actions to be taken in the event of a relief valve malfunction These operator actions are taken to avoid the development of temperatures approaching the 170°F threshold for potential damage by the steam quenching phenomenon.

CONCLUSION

We have concluded, based on the considerations 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.

Date: JUL 1 5 1975

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-263

NORTHERN STATES POWER COMPANY

NOTICE OF PROPOSED ISSUANCE OF AMENDMENT
TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. DPR-22 issued to Northern States Power Company (the licensee), for operation of the Monticello Nuclear Generating Plant (the facility) located in Wright, County, Minnesota.

The amendment would incorporate additional suppression pool water temperature limits: (1) during any testing which adds heat to the pool, (2) at which reactor scram is to be initiated and (3) requiring reactor pressure vessel depressurization. It also would add surveillance requirements for visual examination of the suppression chamber during each refueling and following operations in which the pool temperatures exceed 160°F and add monitoring requirements of water temperatures during operations which add heat to the pool.

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, which are set forth in the proposed license amendment.

By **AUG 25 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

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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 Mr. Gerald Charnoff, Esquire, Shaw, Pittman, Potts and Trowbridge, 910 - 17th Street, N. W., Washington, D. C. 20006, 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.

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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 application for amendment dated March 24, 1975, which is available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at The Environmental Conservation Library, Minneapolis Public Public Library, 300 Nicollet Mall, Minneapolis, Minnesota 55401. The 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 July 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by:
Dennis L. Ziemann

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

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DATE ➤						

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

JUL 0 1 1975

J. Gallo, Chief Hearing Counsel, OELD

BWR TORUS WATER TEMPERATURE LIMITS AND UNILATERAL TECHNICAL SPECIFICATION CHANGES

We have implemented the "BWR Torus Temperature" Technical Specification changes for the "responsive" and "unresponsive" licensees in accordance with the guidelines provided following approval of the lead cases of Nine Mile Point-1 (unresponsive licensee) and Brunswick-2 (responsive licensee). ~~Two cases yet remain to be completed: Monticello and Cooper; however, these will be finished soon.~~

This action had been concurred in by TR, OR, E. Case and you. As you may recall, our June 10 meeting in E. Case's office (attended by J. Carter, G. Lear, you and I) was the occasion for your concurrence with the lead cases, and simultaneously, concurrence with the new approach for "unilateral Tech Spec change" procedures. Jerry Carter was given the task of reducing the latter procedures to a formal policy/procedural statement..

We now understand that you wish to see the individual letters being sent to BWR licensees for amendment of Technical Specifications as was done via letters dated June 13, 1975 for the two lead cases, NMP-1 and Brunswick-2. Therefore, the letters and their enclosures are forwarded herewith for your concurrence and return to OR for dispatch. Also enclosed, for your information, is a list of the responsive/unresponsive licensees to whom this licensing action applies.

Karl R. Goller

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

Enclosures:
1. List of Responsive/Unresponsive Licensees
2. Letters to Licensees

cc: Attached to each action package

AMERICAN REVOLUTION BICENTENNIAL 1776-1976

ENCLOSURE

JUL 0 1 1975

Licensing Action
Technical Specifications Change
BWR Torus Water Temperature Limits

RESPONSIVE LICENSEES	PLANT	DOCKET
Commonwealth Edison Co.	Dresden 2/3	50-237/249
Commonwealth Edison Co.	Quad Cities 1/2	50-254/265
Tennessee Valley Authority**	Browns Ferry 1/2	50-260/296
Northern States Power Co.	Monticello	50-263
Vermont Yankee Nuclear Power Corp.	Vermont Yankee	50-271
Philadelphia Electric Company	Peach Bottom 2/3	50-277/278
Boston Edison Company	Pilgrim	50-293
Iowa Electric Light & Power Co.	Duane Arnold	50-331
Georgia Power Company	Edwin I. Hatch 1	50-321
Carolina Power & Light Co.*	Brunswick-2	50-325

UNRESPONSIVE LICENSEES	PLANT	DOCKET
Jersey Central Power & Light	Oyster Creek	50-219
Niagara Mohawk Power Corp.*	Nine Mile Point-1	50-220
Northeast Nuclear Energy Co.	Millstone Unit 1	50-245
Nebraska Public Power District	Cooper	50-298
Power Authority State of N. Y.	FitzPatrick	50-333

* Lead cases - letters sent 6/13/75

** This change will be implemented in Tech Specs for Browns Ferry 1/2 when they return to operation later this year.