Docket No. 50-293

JUL 15 1975

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Boston Edison Company
ATTN: Mr. Maurice J. Feldmann
Vice President
Operations and Engineering
800 Boylston Street
Boston, Massachusetts 02199

Gentlemen:

The Commission has requested the Federal Register to publish the enclosed Notice of Proposed Issuance of an amendment to Facility License No. DPR-35 for the Pilgrim Nuclear Power Station. The proposed amendment includes a change to the Technical Specifications and is in response to your request dated March 31, 1975, which was submitted in reply to 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 were necessary for clarification and completeness. Your staff disagreed with one modification which requires that the suppression pool temperature be logged every 5 minutes during relief valve operation, but accepted the modifications. 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

a/P

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

OELD
S.A. Treby
7/15/75

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

cc w/enclosures:
See next page

OFFICE		RL:ORB#2	RL:ORB#2	RL:ORB#3	RL:ORB#2	RL:AD/ORs
SURNAME		RMDiggs	PWO'Connor:tc	CJDeBevec	DLZiemann	KRGoller
DATE		6/20/75	6/20/75	6/23/75	6/25/75	7/15/75

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

cc w/enclosures:
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cc w/enclosures and BEC's
filing of 3/31/75:
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Acting Commissioner of Public
Health
Massachusetts Department of
Public Health
600 Washington Street
Boston, Massachusetts 02111

Mr. Wallace Stickney
Environmental Protection Agency
JFK Federal Building
Boston, Massachusetts 02203

BOSTON EDISON COMPANY

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION

PROPOSED AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.
License No. DPR-35

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

   A. The application for amendment by Boston Edison Company (the licensee) dated March 31, 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-35 is hereby amended to read as follows:

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"B. 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

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

Attachment:
Change No. to the
Technical Specifications

Date of Issuance:

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PROPOSED CHANGE TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Delete pages 152, 166 and 167 from the Appendix A Technical Specifications and insert the attached replacement pages 152, 152A, 166 and 167. The changed areas on the revised pages are shown 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 pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water volume and temperature shall be maintained within the following limits except as specified in 3.7.A.2.

   a. Minimum water volume - 84,000 ft^3

   b. Maximum water volume - 94,000 ft^3

   c. Maximum suppression pool temperature during normal continuous power operation shall be $\leq$ 80°F, except as specified in 3.7.A.1.e.

   d. Maximum suppression pool temperature during RCIC, HPCI or ADS operation shall be $\leq$ 90°F, except as specified in 3.7.A.1.e.

   e. In order to continue reactor power operation, the suppression chamber pool temperature must be reduced to $\leq$ 80°F within 24 hours.

   f. If the suppression pool temperature exceeds the limits of Specification 3.7.A.1.d, RCIC, HPCI or ADS testing shall be terminated and suppression pool cooling shall be initiated.

   g. If the suppression pool temperature during reactor power operation exceeds 110°F, the reactor shall be scrammed.

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. a. The suppression chamber water level and temperature shall be checked once per day.

   b. Whenever there is indication of relief valve operation or testing 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.

   c. 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 external visual examination of the suppression chamber shall be conducted before resuming power operation.

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

3.7 CONTAINMENT SYSTEMS (Cont'd)

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

2. Primary containment integrity 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 "open vessel" physics tests at power levels not to exceed 5 Mw(t).

4.7 CONTAINMENT SYSTEMS (Cont'd)

2. Integrated Leak Rate Testing

a. The primary containment integrity shall be demonstrated by performing an Integrated Primary Containment Leak Test (IPCLT) in accordance with either Method A or Method B, as follows:

Method A

Perform leak rate test prior to initial unit operation at the test pressure of 45 psig, P_t (45), to obtain measured leak rate L_m (45), or

Method B

Perform leak rate test prior to initial unit operation at the test pressure of 45 psig, P_t (45), and 23 psig, P_t (23), to obtain the measured leak rates, L_m (45) and L_m (23), respectively.

BASES:

3.7.A & 4.7.A Primary Containment

The integrity of the primary containment and operation of the core standby cooling system in combination limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time, thus greatly reducing the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth such that a rod drop would not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep off-site doses well below 10 CFR 100 limits.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1035 psig. Since all of the gases in the drywell are purged into the pressure supression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 45 psig which is below the maximum of 62 psig. Maximum water volume of 94,000 ft^3 results in a downcomer submergency of 4'9" and the minimum volume of 84,000 ft^3 results in a submergence approximately 12-inches less. The majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate.

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for core standby cooling systems operability as explained in basis 3.5.F.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the pressure 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 pressure suppression chamber loadings.

BASES:

3.7.A & 4.7.A Primary Containment

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.

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 vicinty of the relief valve discharge since these are expected to be the points of highest stress.

If a loss-of-coolant accident were to occur when the reactor water temperature is below approximately 330°F, the containment pressure will not exceed the 62 psig code permissible 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 temperature above 212°F provides additional margin above that available at 330°F.

Inerting

The relatively small containment volume inherent in the GE-BWR pressure suppression containment and the large amount of zirconium in the core are such that the occurrence of a very limited (a percent or so) reaction of the zirconium and steam during a loss-of-coolant accident could lead to the liberation of hydrogen combined with an air atmosphere to result in a flammable concentration in the containment. If a sufficient amount of hydrogen is generated and oxygen is available in stoichiometric quantities, the subsequent ignition of the hydrogen in rapid recombination rate could lead to failure of the containment to maintain a low leakage integrity. The 5% oxygen concentration minimizes the possibility of hydrogen combustion following a loss-of-coolant.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT TO LICENSE NO. DPR-35
AND
CHANGE TO THE TECHNICAL SPECIFICATIONS

SUPPRESSION POOL WATER TEMPERATURE LIMITS

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

INTRODUCTION

By letter dated March 31, 1975, Boston Edison Company (BE) requested a change in the Technical Specifications appended to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station located at Plymouth, Massachusetts. 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 BE 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 Pilgrim 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.

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 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 Pilgrim plant limit the torus pool temperature to 80°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 80°F. While this 80°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 90°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 which are standard operating procedures at Pilgrim. 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 15 1975

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-293

BOSTON EDISON 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-35 issued to Boston Edison Company (the licensee), for operation of the Pilgrim Nuclear Power Station (the facility) located near Plymouth, Massachusetts.

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 August 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 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. Dale G. Stoodley, Counsel, Boston Edison Company, 800 Boylston Street, Boston, Massachusetts 02199, 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 31, 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 Plymouth Public Library, on North Street in Plymouth, Massachusetts 02360. 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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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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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

ENCLOSURE JUL 3 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.