Docket File
DCS-MS-016

MAR 5 1982

Docket No. 50-336

Mr. W. G. Counsil, Senior Vice President
Nuclear Engineering & Operations
Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06101

Dear Mr. Counsil:

The Commission has issued the enclosed Amendment No. 74 to Facility Operating License No. DPR-65 for Millstone Nuclear Power Station, Unit No. 2. This amendment consists of changes to the Technical Specifications (TS) in response to your applications dated December 17, 1981 and January 14, 1982, as supplemented on numerous other dates.

This amendment authorizes Cycle 5 operation at 2700 MWt with:

0 A mixed core with two-thirds Westinghouse (W) fuel and one-third Combustion Engineering (CE) fuel;

0 Modified (sleeved, reduced flow and insert) guide tubes for the control element assemblies; and

0 An additional 704 steam generator tubes plugged.

The amendment revises the Appendix A Technical Specifications by:

0 Incorporating changes resulting from the analysis of the Cycle 5 reload with Westinghouse fuel;

0 Removing the requirements for mid-cycle moderator temperature coefficient determination;

0 Redefining the fully withdrawn position of the regulating control rods; and

0 Modifying the pressurizer level operational band.

cp

A number of the miscellaneous TS changes requested by the December 17, 1981 application were approved by Amendment No. 72, issued on February 22, 1982. This amendment and the previous amendment completes all items of your reference applications.

Some portions of your proposed Technical Specifications have been modified to meet our requirements. These modifications have been discussed with and agreed to by your staff.

OFFICE		8203300274 820305					
SURNAME		PDR ADOCK 05000336					
DATE		P PDR					

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

A copy of the Safety Evaluation and the related Notice of Issuance are also enclosed.

Sincerely,

Original signed by

Eben L. Conner, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 74 to DPR-65
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

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WASHINGTON, D.C. 20555

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Docket No. 50-336

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SUBJECT: NORTHEAST NUCLEAR ENERGY COMPANY, ET AL., Millstone Nuclear Power Station, Unit No. 2

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).

☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.

☐ Notice of Availability of Applicant's Environmental Report.

☐ Notice of Proposed Issuance of Amendment to Facility Operating License.

☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.

☐ Notice of Availability of NRC Draft/Final Environmental Statement.

☐ Notice of Limited Work Authorization.

☐ Notice of Availability of Safety Evaluation Report.

☐ Notice of Issuance of Construction Permit(s).

☐ Notice of Issuance of Facility Operating License(s) or Amendment(s).

☒ Other: Amendment No. 74.
Referenced documents have been provided PDR.

Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE→	ORB#3:DL					
SURNAME→	PMKreutzer/pn					
DATE→	3/5/82					

NRC FORM 102 7—79

Northeast Nuclear Energy Company

cc:

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cc w/enclosure(s) and incoming
dtd: 12/17/81, 1/14/82

Office of Policy & Management
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NORTHEAST NUCLEAR ENERGY COMPANY

THE CONNECTICUT LIGHT AND POWER COMPANY

THE HARTFORD ELECTRIC LIGHT COMPANY

THE WESTERN MASSACHUSETTS ELECTRIC COMPANY

DOCKET NO. 50-336

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 74
License No. DPR-65

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

   A. The applications for amendment by Northeast Nuclear Energy Company (the licensee) dated December 17, 1981 and January 14, 1982 as supplemented, comply 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 applications, 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-65 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 74 , 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

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: MAR 5 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 74

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

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B 3/4 1-1

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be $\geq$ 2.90% $\Delta k/k$.

APPLICABILITY: MODES 1, 2*, 3 and 4.

ACTION:

With the SHUTDOWN MARGIN < 2.90% $\Delta k/k$, within 15 minutes initiate and continue boration at $\geq$ 40 gpm of boric acid solution at or greater than the required refueling water storage tank (RWST) concentration (ppm) until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be $\geq$ 2.90% $\Delta k/k$:

a. Immediately upon detection of an inoperable CEA. If the inoperable CEA is immovable or untrippable, the SHUTDOWN MARGIN, required by Specification 3.1.1.1, shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA.

b. When in MODES 1 or 2, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. Prior to initial operation above 5% RATED THERMAL POWER after each refueling, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

*See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

d. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:

    1. Reactor coolant system boron concentration,
    2. CEA position,
    3. Reactor coolant temperature,
    4. Fuel burnup based on gross thermal energy generation,
    5. Xenon concentration, and
    6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% Δk/k at least once per 31 Effective Full Power Days. This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.d, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each refueling.

MODERATOR TEMPERATURE COEFFICIENT (MTC)

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.5×10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER,

b. Less positive than 0.4×10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is > 70% of RATED THERMAL POWER, and

c. Less negative than -2.4×10^{-4} $\Delta k/k/°F$ at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the predicted values.

* With $K_{eff} \geq 1.0$.

See Special Test Exemption 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each refueling.

b. At any THERMAL POWER, within 14 EFPD after each fuel loading at equilibrium boron concentration.

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to at least 176 steps.

APPLICABILITY: MODES 1 and 2*.

ACTION:

With a maximum of one shutdown CEA withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, to less than 176 steps, either:

a. Withdraw the CEA to at least 176 steps within one hour, or

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to at least 176 steps:

a. Prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exception 3.10.2.

REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits shown on Figure 3.1-2. Regulating CEAs are considered to be fully withdrawn in accordance with figure 3.1-2 when withdrawn to at least 176 steps. With CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits restricted to:

a. $\leq$ 4 hours per 24 hour interval,

b. $\leq$ 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

c. $\leq$ 14 Effective Full Power Days per calendar year.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

a. With the regulating CEA groups inserted beyond the Transient Insertion Limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours either:

1. Restore the regulating CEA groups to within the limits, or

2. Reduce THERMAL POWER to that fraction of RATED THERMAL POWER which is allowed by the CEA group position using the above figures.

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals > 4 hours per 24 hour interval, except during operation pursuant to the provisions of ACTION items c. and d. of Specification 3.1.3.1, operation may proceed provided either:

1. The Short Term Steady State Insertion Limits of Figure 3.1-2 are not exceeded, or

2. Any subsequent increase in THERMAL POWER is restricted to $\leq$ 5% of RATED THERMAL POWER per hour.

*See Special Test Exception 3.10.2 and 3.10.5.

#With $K_{eff} \geq 1.0$.

RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.3 Two power operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one or more PORV(s) inoperable, within 8 hours either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more block valve(s) inoperable, within 8 hours either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.3.1 Each PORV shall be demonstrated OPERABLE:

a. Once per 31 days by performance of a CHANNEL FUNCTIONAL TEST, excluding valve operation, and

b. Once per 18 months by performance of a CHANNEL CALIBRATION.

4.4.3.2 Each block valve shall be demonstrated OPERABLE once per 92 days by operating the valve through one complete cycle of full travel. This demonstration is not required if a PORV block valve is closed and power removed to meet Specification 3.4.3 a or b.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a steam bubble and with at least 130 kw of pressurizer heater capacity capable of being supplied by emergency power. The pressurizer level shall be within ± 5% of its programmed value during periods of normal operation.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

A. With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

B. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4 The pressurizer water level shall be determined to be within ± 5% of its programmed value at least once per 12 hours.

* During transient operations (startup, power level changes, trips, etc.) the pressurizer level may be outside the ± 5% band for periods not to exceed one hour.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 2.90%$\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN required by Specification 3.1.1.1 is based upon this limiting condition and is consistent with FSAR accident analysis assumptions. For earlier periods during the fuel cycle, this value is conservative. With $T_{avg} \leq 200°F$, the reactivity transients resulting from any postulated accident are minimal and a 2% $\Delta k/k$ shutdown margin provides adequate protection.

3/4.1.1.3 BORON DILUTION AND ADDITION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration changes in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 10,060 + 700/-0 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron concentration changes will be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurance that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

The MTC is expected to be slightly negative at operating conditions. However, at the beginning of the fuel cycle, the MTC may be slightly positive at operating conditions and since it will become more positive at lower temperatures, this specification is provided to restrict reactor operation when T_{avg} is significantly below the normal operating temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 1.0% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 4550 gallons of 6.25% boric acid solution from the boric acid tanks or 47,300 gallons of 1720 ppm borated water from the refueling water storage tank.

The requirements for a minimum contained volume of 370,000 gallons of borated water in the refueling water storage tank ensures the capability for borating the RCS to the desired level. The specified quantity of borated water is consistent with the ECCS requirements of Specification 3.5.4. Therefore, the larger volume of borated water is specified here too.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 74 TO FACILITY OPERATING LICENSE NO. DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

TABLE OF CONTENTS

1.0 Introduction

By applications dated December 2 and 17, 1981 and January 14, 1982 (Ref. 811202, 811217 and 820114)* and supplemental information as listed in the reference sections, Northeast Nuclear Energy Company (NNECO or the licensee) requested an amendment ot Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Unit No. 2 (Millstone-2 or the facility). The amendment request consists of:

- Appendix A (Safety) Technical Specifications (TS) changes resulting from the analyses of the Cycle 5 reload fuel;
- Continued approval to operate with modified (sleeved, reduced flow and insert) Control Element Assembly (CEA) guide tubes;
- Approval to operate with an additional 704 steam generator tubes plugged; and
- Evaluation of numerous changes partially ~~related~~ to Cycle 5 operation.

The specific request of the December 2, 1981 application, to modify the operability requirements for two independent shutdown cooling loops, was issued by Amendment No. 71 (Ref. 811218). To simplify this reload Safety Evaluation (SE), numerous other changes partially related to Cycle 5 operation were issued by Amendment No. 72 (Ref. 820222). The steam generator (SG) tube pitting and resultant inspection program and tube plugging is addressed in the SE supporting this Amendment.

The associated specific TS changes are described in Section 3.0 of the following SE.

In early 1977, NNECO indicated to the NRC staff their intention to change fuel assembly vendors from Combustion Engineering, Inc. (CE) to Westinghouse Electric Corporation (W). In March 1980, NNECO submitted the Basic Safety Report (BSR), (Ref. 800306) authorized by W for Millstone-2. This BSR in part supersedes the original FSAR that was prepared by CE. Our evaluation and approval of the BSR is given in References 810622, 820112 and 820218. Our evaluation of the Cycle 5 reload safety analysis (RSA) will not address those issues (e.g., Westinghouse reload fuel design bases, rod bowing analyses, etc.) which were resolved in our above referenced approvals of the BSR.

*Reference number made up of year, month and day in that order.

2.0 Discussion and Evaluation

In this evaluation of the Cycle 5 reload using, for the second cycle, fuel assemblies designed and manufactured by Westinghouse in the Millstone-2 core, use is made of our generic review of the BSR and various other topical reports. Some of the topical reports have not received formal NRC staff approval. In all cases where a topical report has not received such an approval, the report has been examined, its methods judged to be reasonable, and an appraisal has been made that a complete review will not reveal the methodology to be significantly in error. On this basis, all topicals referenced are judged to be acceptable for this reload of Millstone-2 and for operation at the licensed power level of 2700 MWt.

2.1 Fuel System Design

The objectives of the fuel system safety review are to provide assurance that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained. We have reviewed the information provided in support of Millstone-2, Cycle 5 operation to determine if these objectives have been met.

The Millstone-2, Cycle 5 core will be comprised of (a) 73 fuel assemblies that were manufactured by Combustion Engineering, the original NSSS vendor, and (b) 144 fuel assemblies supplied by Westinghouse, the Cycle 4 and 5 reload fuel vendor. The Cycle 5 core loading inventory is given in the following table.

Millstone, Unit No. 2, Cycle 5 Core Loading Inventory

Assembly Designation	Vendor	Number of Assemblies	Initial Enrichment (w/o U235)	Theoretical Density (%)	BOC Assembly Average Exposure (MWD/MTU)
B+	CE	1	2.336	95	17,450
E1	CE	24	2.730	94.75	24,650
E2	CE	48	3.235	94.75	22,600
F1	W	24	2.697	94.54	13,470
F2	W	48	3.297	94.87	9,650
G1	W	24	2.70	95*	0
G2	W	48	3.20	95*	0
		217 total			

*Region G1 and G2 densities are nominal. Average densities of 94.5% were used in the safety analysis.

The fuel management pattern was developed to accommodate a Cycle 4 burnup range of 10,650 MWD/MTU to 12,000 MWD/MTU. After the core reload, the beginning-of-cycle core-average exposure will be about 11,430 MWD/MTU making the predicted end-of-cycle core-average exposure about 21,830 MWD/MTU (Ref. 811221).

The Westinghouse reload fuel was designed to be geometrically similar to and compatible with the Combustion Engineering reference fuel. Table 1 of Reference 820218 provides a comparison of the fuel mechanical designs.

2.1.1 Seismic-and-LOCA Mechanical Response

As discussed in the Millstone-2 Cycle 4 reload SER (Ref. 801006) and the BSR SER (Ref. 820218), both CE and W performed analyses of the fuel response to combined seismic-and-LOCA loadings. Each of those analyses was performed for a homogeneous core of one type of fuel (e.g., CE or W). Because Cycle 5 operation of Millstone-2 will involve a heterogeneous core of both CE and W fuel, a mixed-core seismic-and LOCA analysis was required.

The licensee submitted a mixed-core analysis (Ref. 810501 and 810608), which shows that (a) the maximum deformation occurs in peripheral W assemblies and (b) this deformation does not invalidate the results of the current LOCA analysis.

It should be noted that mechanical response analyses have not been completely reviewed at this time, but that the Task Action Plan for this generic issue (Ref. 781100) provides a basis for continued operation while the issue is being fully resolved. Because of (a) the present unreviewed status of the underlying primary systems asymmetric loads analysis, (b) the temporary existence of a mixed core in Millstone-2, (c) our previous approval of the W analytical methods for the fuel assembly response, and (d) the favorable analytical result reported by the licensee, we consider the mixed-core issue to be adequately resolved for Cycle 5 operation without further review.

2.1.2 CEA and Fuel Assembly Guide Tube Wear

Background information on fretting wear of CEA (control element assembly) cladding and fuel assembly guide tubes can be found in the safety evaluation reports (Ref. 820218 and 801006) that were written on the Millstone-2, BSR and the Cycle 4 safety analysis report, respectively.

In order to provide continuing assurance of both CEA and guide tube integrity, NNECO has outlined (Ref. 810928) a proposed surveillance program to be performed following Cycle 4 operation. The program will involve a combination of visual, profilometric, and eddy current examinations of 2 to 6 CEAs and 6 to 16 fuel assemblies. The fuel assemblies to be examined will include both CE and W standard sleeved assemblies and demonstration assemblies that were all positioned in rodded core locations during Cycle 4 operation. The demonstration assemblies employed in Cycle 4 consisted of 4 CE assemblies of the reduced guide tube flow design and 4 W assemblies having guide tube inserts.

We believe that the NNECO surveillance techniques and proposed program will be adequate for establishing CEA and guide tube integrity. NNECO has agreed to formally submit the examination results for NRC review within 90 days following Millstone-2 restart. In that submittal, we recommend that NNECO describe plans for continuing CEA and guide tube surveillance or provide justification for discontinuing those specific examinations.

On the basis of an anticipation of acceptable wear measurements from the surveillance program and the fact that all rodded Cycle 5 fuel assemblies (except 4 W assemblies with guide tube inserts) will be sleeved, we conclude that NNECO has provided sufficient justification for Cycle 5 operation.

2.1.3 Cladding Collapse

As described in the safety evaluation on the BSR, individual reload applications referencing the cladding creep-collapse analysis of W reload fuel should confirm that the collapse analysis was performed in accordance with the condition of approval placed on the W generic analytical method. That condition involves the use of specified input curves (e.g., initial ovality) for the analysis.

The licensee has stated (Ref. 820204) that the input curves were used as specified by the SER and that the W reload fuel is not predicted to collapse during Cycle 5 operation. Hence, this issue is satisfied.

The licensee has completed (Ref. 820223) the Cycle 5 cladding collapse analysis for the CE fuel. CE fuel is pressurized to preclude cladding collapse and analyzed with conservative methods that demonstrate free-standing cladding beyond a 34,500 EFPH exposure, which bounds the lead fuel rod exposure for Cycle 5. Therefore, we conclude that no cladding collapse will occur during Cycle 5.

2.1.4 Fuel Manufacturing Problems

Dimensional checks to ensure that the reload fuel assemblies are compatible with other core components are a part of the new-fuel receipt inspection program performed by NNECO at Millstone-2. The inspection of Cycle 5 reload fuel revealed 2 conditions which required a more thorough examination and which resulted in the need to ship some fuel assemblies back to the Columbia, South Carolina fuel fabrication facility for modifications (see Ref. 820108). The required modifications to the Cycle 5 reload fuel assemblies were not as extensive and unlike those that were previously required for all of the W Cycle 4 reload fuel assemblies (see Ref. 801006).

The first condition which indicated a problem was discovered during the envelope inspection of fuel assembly top nozzles. An onsite upper gauge block (UGB), which is designed to seat on top nozzle posts, is used to verify that fuel assemblies will align properly under the upper core plate. The UGB would not seat on some fuel assemblies due to one or both of the following: (1) top nozzle plates were not parallel to bottom nozzle plates or (2) top nozzle posts were misaligned or irregularly spaced. Since the UGB is built to require closer seating tolerance than the core plate, W designed and built a gauge block which more closely represented the tolerance needed for the Millstone-2 core plate. All but 4 fuel assemblies passed the inspection with the W gauge block. These 4 assemblies were subsequently modified at the fuel fabrication facility.

The second condition that indicated a problem was encountered during the CEA free-path and end-clearance checks. The licensee reported that most fuel assemblies failed this examination; consequently, measurements of available CEA lateral clearance at the bottom of guide tubes were taken on each fuel assembly. The reduction in clearances in the lowermost portion of the guide tubes is attributable to the use of guide tube end plugs which were left over from the Cycle 4 fuel production lot. When these end plugs were swaged to the bottom of guide tubes, the diameter of the guide tube walls was locally crimped. Fortunately, NNECO's measurements revealed that adequate clearance for CEA operation remained for all fuel assemblies and for all instances of worst-case conditions. The Cycle 6 reload fuel will employ a different end plug design.

W is considering fabrication modifications that will preclude similar occurrences in the future, and NRC's Office of Inspection and Enforcement is reviewing quality assurance controls that are used at the fuel fabrication facility. We conclude that both Cycle 4 and Cycle 5 reload fuel fabrication problems arose because of W's inaccessibility to CE proprietary information on the design of Millstone-2 fuel. Inasmuch as W has now supplied 2 reloads for Millstone-2, we would not anticipate further problems of this nature in the future.

2.1.5 Miscellaneous Analyses

We asked NNECO about 2 issues that were not addressed in the reload safety analysis report. Those issues were supplemental ECCS calculations with the cladding models of NUREG-0630 and fuel rod bowing analyses for CE fuel. The licensee stated (Ref. 820204) that the analyses of these issues that were performed for Cycle 4 operation are bounding relative to those for the planned Cycle 5 operation of Millstone-2. We accept this response without further question.

2.2 Nuclear Design

The nuclear design procedures and models used for the analysis of the Millstone-2, Cycle 5 reload core are the same as those used for Cycle 4. These are documented in the Millstone-2 BSR and have been approved for the analysis of the Millstone-2 core using W reload fuel beginning with Cycle 4.

2.2.1 Control Rod Worth

The control rod worths and shutdown requirements for both beginning and end-of-cycle (EOC) 5 are presented and compared with previous Cycle 4 values. At EOC 5, the reactivity worth with all control rods inserted assuming the highest worth rod is stuck out of the core is 5.93% $\Delta\rho$ assuming a 10% uncertainty reduction. The reactivity worth required for shutdown, including the contribution required to control the steam line break even at EOC 5 is 5.90% $\Delta\rho$. Therefore, sufficient control rod worth is available to accommodate the reactivity effects of the steam line break at the worst time in core life allowing for the most reactive control rod stuck in the fully withdrawn position and also allowing for calculational uncertainties in these worths based upon comparison of calculations with experiments presented in the BSR and in previous W reports. On the basis of our review, we have concluded that NNECO's assessment of reactivity control is suitably conservative and that adequate negative reactivity worth has been provided by the control system to assure shutdown capability assuming the next most reactive control rod is stuck in the fully withdrawn position.

2.2.2 Moderator and Doppler Temperature Coefficients

The most positive moderator temperature coefficient between 70% to 100% power has increased to $+0.4 \times 10^{-4}$ $\Delta\rho/°F$ from the Cycle 4 value of $+0.2 \times 10^{-4}$ $\Delta\rho/°F$. The Doppler coefficient has been extended to -1.92×10^{-5} $\Delta\rho/°F$ compared to the Cycle 4 most negative value of -1.87×10^{-5} $\Delta\rho/°F$. The maximum delayed neutron fraction has also increased slightly from the previous cycle value. The maximum differential rod worth of two CEA groups moving together (at hot zero power) has increased from 24.3×10^{-5} $\Delta\rho$/in to $36.6 \times 10^{-5} \Delta\rho$ /in. These changes, as well as changes in the total trip reactivity as a function of position and the Doppler power coefficient as a function of power, exceed the limiting range of values established by the Cycle 4 and BSR safety analysis. Therefore, reanalyses of those transients which are affected by these kinetics parameters were performed (see Section 2.4 of this SE).

2.3 Thermal-hydraulic Design

The thermal-hydraulic design for Millstone-2 is presented in the BSR (Ref. 800306).

2.3.1 Hydraulic Compatibility

As discussed in the BSR, the W Cycle 5 reload fuel assemblies for Millstone-2 are designed, and shown through testing, to be hydraulically compatible with the CE Cycle 3 reference fuel assemblies.

2.3.2 Design Power Level

The design power level for Millstone-2, Cycle 5 remains 2700 MWt (the same as for Cycle 4). The safety analysis uses a power level of 2754 MWt (102% power) to allow for measurement uncertainties. A summary of our evaluation follows.

2.3.3 Analytical Methods

The steady state DNB analysis for Cycle 5 was performed using the THINC-I code in conjunction with the W-3 correlation (Refs. 7803, 72, 6901 and 7201). For the W-3 correlation, the 95/95 confidence/probability limit for not suffering departure from nucleate boiling is a DNBR greater than 1.30. In the analysis, uncertainties in various measured parameters were factored in as biases for LCO and LSSS setpoints. This biasing of the measurements uncertainties in the analysis is equivalent to adding the absolute power uncertainties in the various measured parameters and applying the total power uncertainty to the best estimate calculation. The specific uncertainties along with their equivalent power uncertainties for Cycles 4 and 5, as determined with the THINC-I code in conjunction with the W-3 correlation (grid spacer correction = 1.0), and for Cycle 3, as determined with the TORC thermal hydraulic code in conjunction with the CE-1 correlation, are as follows.

Percent Uncertainties

Measured Parameter	Measured Parameter Uncertainty	Equivalent Power Uncertainty - %		
		Cycle 3	Cycle 4	Cycle 5
Axial Shape Index (ASI)	0.06 ASIU	2.2%	3.0%	3.0%
Pressure	22 psi	0.8%	0.5	0.5
Temperature	2°F	0.9	1.0	1.0
Flow	4%	5.0	2.0	2.0
Power (LCO)	2%	1.4	2.0	2.0
Power (LSSS)	5%	3.5	5.0	5.0

NOTE: Cycle 3 determined with TORC code in conjunction with CE-1 correlation.
Cycle 4 determined with THINC-1 code in conjunction with W-3 correlation.

LCO = Limiting Conditions for Operation
LSSS = Limiting Safety Systems Settings

The following parameters related to LCO and LSSS are the same for Cycles 3, 4 and 5: power level (2754 MWt), maximum steady state core inlet temperature (551°F), minimum reactor coolant flow (133.7 x 10^6 lb/hr), and maximum allowed initial peak linear heat rate (16.0 kw/ft).

NNECO agreed to provide justification for the measurement uncertainty values (Axial Shape Index (ASI), Pressure, Temperature, Flow Power (LCO) and Power (LSSS)) for further review of the Cycle 4 and Cycle 5 power uncertainties. This will be supplied by March 1, 1982. While our review of measurement uncertainties continues, LCO and LSSS limits for Cycle 5 will be maintained at the values used for Cycle 3 (Ref. 800603). We find this acceptable.

2.3.4 Reactor Coolant Flow

The design flow for the Cycle 5 analysis is 370,000 gpm (133.7 x 10^6 lb/hr at 2200 psi and 551°F) and is the same as the low flow limit included in the TS and analysis for Cycle 4.

2.3.5 Limiting Transient-Complete Loss of Reactor Coolant Flow

The loss of flow accident was reanalyzed for Cycle 5. The results show that the reactor trip protection provided by the reactor coolant pump speed sensing system is sufficient to prevent cladding and fuel damage. The DNBR aproaches but does not decrease below 1.30 during the transient. This is the same result as for Cycle 4. We find this acceptable.

2.4 Accident Analyses

The licensee's analysis of accidents for Cycle 4 was provided in the BSR (Ref. 800306) and the Cycle 4 RSA (Ref. 800603). Our approval for Cycle 4 operation (Ref. 801006) found these accident analyses acceptable. For Cycle 5, NNECO has reanalyzed the boron dilution event-Modes 2, 3 and 4, the CEA ejection event, the CEA withdrawal event-Modes 1 and 2, the complete loss of reactor coolant flow event and the RCP seized rotor event. The Cycle 5 RSA states that this reanalysis was necessary because of changes in cycle-specific parameters in the area of kinetic characteristics, CEA worths, and core peaking factors (Ref. 811117). We find that the correct reanalyses of accidents have been performed.

By References 820204 and 820301, NNECO provided the results of their review to determine the acceptability of transient and accident analyses considering the increase of plugged steam generator tubes from 500 to 750 per generator. They concluded that the resultant change in RCS flow and heat transfer characteristics do not change the previously docketed non-LOCA transient and accident analyses. We have reviewed their presentation and concur with their findings.

The reanalyzed accidents are evaluated as follows.

2.4.1 Boron Dilution Event

An inadvertent boron dilution will reduce the boron concentration in the primary coolant which in turn will increase the reactor core positive reactivity. During power operation, the resulting reactivity insertion will increase the reactor power and automatic safety systems will act to shut down the reactor and maintain the plant within safety limits. However, a boron dilution event during shutdown will not be mitigated by any automatic safety systems. It may continue and result in reactor criticality if the operator does not take the appropriate corrective action within the necessary time period.

In Reference 811117, NNECO indicates that the shutdown margin requirements for Cycle 5 are more limiting than those for Cycle 4 for Modes 1, 2, 3 and 4. Therefore, the operator action time available before a complete loss of shutdown margin occurs during a boron dilution event is less in these modes for Cycle 5. The transient was reanalyzed for these hot modes (startup, hot standby, and hot shutdown) only. It is not necessary for power operation (Mode 1) as discussed above.

The results of the reanalyses were 64, 24 and 24 minutes to lose shutdown margin for the startup, hot shutdown and hot standby modes, respectively. The reduction in these times is due to the decreased Modes 1, 2, 3 and 4 shutdown margin from 3.20 to 2.90. This decrease results from the Cycle 5 specific analyses. We find these results exceed our 15 minute criteria and are, therefore, acceptable.

Since the shutdown margin for Mode 5 (cold shutdown) and Mode 6 (refueling) has not changed, NNECO did not reanalyze the boron dilution event for these modes. The staff did, however, request additional information regarding the ability of the installed instrumentation channels to detect and alert the operator of a boron dilution event and the resultant operator action time available (Ref. 811224). The information has been provided (Ref. 820204). Our finding in the Amendment 61 Cycle 4 reload SE (Ref. 801006) was, "The limiting dilution event for the Cycle 4 operation is for the refueling mode with a calculated time to criticality of 34 minutes which is more than the required 30 minutes. Therefore, we find this analysis and its results acceptable for all cases when the reactor is subcritical." We now conclude that this finding is also applicable to Cycle 5 operations. However, the licensee should be aware that the staff is presently evaluating the need for all operating PWRs to provide additional protection against uncontrolled boron dilution events during the shutdown modes.

Pending the outcome of this evaluation, it may be necessary to require additional instrumentation to alert the operator of a boron dilution event. NNECO will be notified if any such action is necessary.

2.4.2 <u>CEA Ejection Incident</u>

The more positive moderator temperature coefficent, between 70% and 100% power, required a reanalysis of the CEA ejection incident initiated from hot full power conditions. The higher total peaking factor after ejection (compared to the BSR value) required a reanalysis of the hot zero power CEA ejection incident. The results indicated that the Regulatory Guide 1.77 limiting criterion of 280 cal/gm is not exceeded for either case. We have reviewed the analysis assumptions including the Doppler and moderator coefficients, delayed neutron fractions, ejected rod worths, hot channel factors and trip reactivity insertion and find the analysis to be conservative and the predicted consequences accetable.

2.4.3 CEA Withdrawal from Subcritical

Changes in the Cycle 5 trip reactivity curve, delayed neutron fraction, and Doppler power coefficient, as well as in the maximum differential rod worth of two CEA groups moving together at hot zero power, required a reanalysis of the CEA withdrawal incident from a subcritical condition. The results show that the DNBR is greater than the limiting value of 1.30 and, therefore, no cladding damage or fission product release to the reactor coolant system will result.

2.4.4 CEA Withdrawal at Power

Changes in the Cycle 5 trip reactivity curve, delayed neutron fraction, and Doppler power coefficient, as well as in the maximum differential rod worth of two CEA groups moving together, required a reanalysis of the CEA withdrawal incident from power. The results show that the thermal margin low pressure trip provides protection over the full range of reactivity insertion rates from 0 to 2.44×10^{-4} $\Delta\rho$/sec so that the minimum DNBR remains above 1.30. We find the CEA withdrawal analyses and consequences accceptable.

2.4.5 Complete Loss of Reactor Coolant Flow

A loss of reactor coolant flow could result from mechanical or electrical failure in one or more of the reactor coolant pumps. The immediate effect of reduced coolant flow is a rapid increase in coolant temperature. This heat up in coolant temperature could lead to DNB and subsequent fuel damage if proper protection were not provided.

The lose of flow accident was reanalyzed for Cycle 5 because of the change in the trip reactivity curve for Cycle 5 and changes in the delayed neutron fraction and Doppler and temperature coefficient. Millstone-2 has provided the following protection against this event:

1. Reactor coolant pump speed sensing system.
2. Low reactor coolant loop flow trip.

The reactor coolant speed sensing system is provided to protect against loss of power to all pumps. The low reactor coolant loop flow trip is provided to protect for loss of one or two reactor coolant pumps.

The licensee has analyzed the transient with three digital computer codes, i.e., LOFTRAN, FRACTRAN AND THINC. The acceptability of these codes is discussed in Reference 820112. The results provided in Reference 811117 indicate that for the most limiting loss of flow event the DNBR decreases to a minimum value of 1.31 at approximately 3.7 seconds into the transient. Core flow at the time of minimum DNBR is approximately 65% of normal full flow. Although there is a turbine-generator assist feature which would provide a slower coastdown, it was not considered for this analysis.

The licensee has performed an analysis of complete loss of coolant flow transient to determine its impact on the DNBR. The results of the analysis indicate that the DNBR does not decrease below 1.30 during the transient. The results also confirm that the analysis as presented in Reference 800306, Millstone Basic Safety Report, continues to bound Cycle 5 plant operation. We, therefore, conclude that the results of this analysis are acceptable.

2.4.6 Reactor Coolant Pump Seized Rotor

The seized rotor transient was reanalyzed for Cycle 5 because of the change in the trip reactivity curve as a function of rod position and other changes in the delay neutron fraction and Doppler and temperature coefficients. The accident postulated is an instantaneous seizure of a reactor coolant pump rotor. Flow through the affected reactor coolant loop is rapidly reduced, leading to an intiation of a reactor trip on a low flow signal.

In Reference 811117, the licensee provided the results of the analysis to demonstrate that the integrity of the primary coolant system would not be endangered since the peak reactor coolant system pressure, approximately 2500 psia, is less than 110% of the RCS design pressure. The peak clad temperature of approximately 1960°F is much less than 2700°F (Millstone-2 fuel design temperature limit) which guarantees that the core will remain intact with no loss of core cooling capabiltiy following the accident. The results also indicate that less than 2 percent of the fuel rods are predicted to experience departure from nucleate boiling.

Since the Cycle 5 plant response to a reactor coolant pump seized rotor transient is within the reactor coolant system pressure and fuel limits, we conclude the results are acceptable.

2.4.7 Steam Line Rupture Accident

The steam line rupture accident was reanalyzed for Cycle 5 because of the change in shutdown margin, trip reactivity curve, and kinetics coefficient. This transient is the most limiting case which assumes the steam line rupture inside the containment at the outlet of the steam generator. The plant initially is at no load condition with offsite power available. The analysis was performed with the assumption that auxiliary feedwater (AFW) flow would be initiated automatically during the transient. It was assumed that 2800 gpm of AFW, 35% more than the maximum runout flow, would be delivered to the affected steam generator at three minutes after the beginning of the transient. This is conservative with respect to the expected time of AFW initiation since automatic actuation of the AFW system would occur on a low steam generator water level trip signal. The assumption was also made that the minimum capabiltiy for injection of boric acid solution (1720 ppm) corresponds to the most restrictive single failure in the safety injection pump and one low pressure safety injection pump delivering full flow to cold leg header. Results provided in Reference 811217 show that the reactor core returns to critical after CEA insertion (assuming the most reactive CEA is stuck in the withdrawn position). This is due to the high cooldown rate, resulting from the steam discharge and auxiliary feedwater addition in the presence of a negative moderator temperature coefficient. However the addition of boron from the high pressure safety injection pump brings the core subcritical again. The peak heat flux attained during this transient is small, approximately 3 percent, and the DNBR margin design basis of 1.30 will not be violated. The maximum pressure within the reactor coolant boundary and the main steam system would not exceed 110 percent of the design pressure. We conclude that appropriate analysis has been provided for this transient and the results of the analysis are acceptable.

2.5 Loss of Coolant Accident

By letter dated February 19, 1982 (Ref. 820219), NNECO provided the LOCA analysis with additional plugged steam generator tubes. The analysis was performed with the approved version of the W evaluation model (1981) assuming 102% licensed core power rating and with 9.4% steam generator tubes (800 tubes per steam generator) plugged. Modification to computer input included reduction in the primary steam generator flow area and volume. One large break calculation is appropriate for this type of reanalysis. The model changes properly reflected changes in plant conditions. We find this reanalysis has been performed in accordance with 10 CFR 50 Appendix K and related staff positions and, therefore, the revised large break LOCA analysis is acceptable.

NNECO states (Ref. 820204-2) that the small break LOCA analysis was reviewed and it has been determined that input parameters were assumed for each steam generator which were equivalent to having approximately 1000 plugged tubes per steam generator. Therefore, they find the current small break LOCA analysis results remain valid for Cycle 5. We concur in this finding.

2.6 Radiological Consequences of Postulated Accidents

We have reviewed the BSR, RSA and the other submittals supporting Cycle 5 operation and find the potential radiological consequences of design basis accidents to be appropriately bounded by the original May 10, 1974 Safety Evaluation or by the Cycle 3 Reload Safety Evalution. Since the guidelines of 10 CFR Part 100 continue to be met, we find the potential consequences acceptable.

3.0 Technical Specification Changes

NNECO proposed the TS changes necessary for Cycle 5 operation in References 811217 and 820114. A large number of the proposed changes not specifically related to the reload were issued by Amendment No. 72 (Ref. 820222). As stated in earlier portions of this SE, the majority of the Cycle 5 analysis using W fuel is, as was the case for Cycle 4, bounded by the Cycle 3 analysis.

where CE fuel was reloaded. The necessary TS changes are as follows.

3.1 Shutdown Margin

NNECO proposed a change in the Modes 1 through 4 shutdown margin from 3.2% $\Delta K/K$ (Cycle 4) to 2.9% $\Delta K/K$ for Cycle 5 (Ref. 820114). This value is a direct result of the cycle specific analyzed core characteristic. We find this change justified by the analysis. The TS pages affected are 3/4 1-1 and B 3/4 1-1.

3.2 Moderator Temperature Coefficient

Reference 811217 Item No. 7 proposes to change the moderator temperature coefficient (MTC) limit in TS 3.1.1.4 from less positive than 0.2×10^{-4} to 0.4×10^{-4} $\Delta K/K/°F$ whenever thermal power is > 70% of rated. This change is the direct result of the Cycle 5 core characteristics and is supported by normal and accident analyses. Therefore, we find this proposed change to Page 3/4 1-5 acceptable.

Item 8 of the same reference requests removal of TS surveillance requirement (SR) 4.1.1.4.2c. This SR is to perform a MTC determination at mid-cycle. NNECO stated that they have successfully demonstrated their capability to predict the MTC through four fuel cycles with two fuel vendors (CE and W). They also contend that: (1) the MTC measurement at beginning of cycle ensures that no unforeseen core characteristics exist; (2) MTC testing is a high risk plant test involving significant CEA movement and axial shape index shifts; and (3) the MTC test requires a loss of production of about 6% for approximately 4 days. We find that the MTC determination at mid-cycle is of marginal value and therefore, because of the adverse aspects of such a test, should be deleted from Page 3/4 1-6.

3.3 Withdrawn Position of Regulating CEAs

The current TS 3.1.3.6 requires that regulating CEAs shall be limited to the withdrawal sequence show on Figure 3.1-2, the CEA insertion limit. NNECO's Reference 811217 proposal is to clarify this specification by indicating that the fully withdrawn position is greater than or equal to 176 steps. This is the approved fully withdrawn position of the shutdown CEAs (see TS 3.1.3.5). We find the proposed change to TS Page 3/4 1-28 will have no adverse effect on reactivity insertion or peaking factors and is, therefore, acceptable.

3.4 Pressurizer Level Control

Amendment No. 66 (Ref. 810407) changed numerous TS pages including Page 3/4 4-4 to implement TMI Category A requirements. The change to Page 3/4 4-4 was to require at least 130kW of pressurizer heater capacity and level within +5% of its programmed value. This last change has proved to be impractical since pressurizer level control during transient operation (startup, power level changes, trips, etc.) is not that precise. NNECO has proposed (Ref. 811217 Item 11) to footnote an exception for level control during transient operations. Although this is one way to correct the TS, we do not believe it is the best. Current STS give only a minimum acceptable level (enough water to prevent heater damage) and a maximum level (enough steam to prevent solid conditions). In discussions with NNECO, they agree that the new STS could provide a better basis, and have initiated their procedure to propose such a TS change including the appropriate STS surveillance requirements. However, since this action will take time and TS violations may occur during this time period, both the licensee and the staff agree to the proposed TS modification to Section 3.4.4 until NNECO can make another proposal. We find the proposed changes to TS Page 3/4 4-4 acceptable.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: MAR 5 1982

Principal Contributors:

Dale Powers	Vince Leung
Harry Balukjian	Norm Lauben
Larry Kopp	Monte Conner

6.0 References

The following references are listed in the chronological order of the date of the transmittal letter. The first two digits are the year, the next two the month and the last two the day of the month.

Reference No.	Description
6901	Westinghouse Report, WCAP-7015, Subchannel Thermal Analysis of Rod Bundle Cores, H. Chelemer, J. Weisman and L. Tong, January 1969.
7200	Commission Report Boiling Crisis and Critical Heat Flux, L. Tong, 1972.
7201	Westinghouse Report, WCAP-7838, Application of the THINC Program to PWR Design, J. Shefcheck, January 1972.
7803	Westinghouse Report, WCAP-9272, Reload Safety Evaluation Methodology, F. Bordelon, March 1978.
7811	NRC Report, NUREG-0371, Task Action Plan for Generic Activities - Category A, November 1978.
800306	NNECO Letter transmitting Westinghouse Basic Safety Report (BSR), W. Counsil to R. Reid, March 6, 1980.
800603	NNECO letter transmitting Cycle 4 Refueling Safety Analysis, W. Counsil to R. Clark, June 3, 1980.
801006	NRC letter transmitting Amendment No. 61, Cycle 4 Reload Evaluation, R. Clark to W. Counsil, October 6, 1980.
810108	NNECO letter, Resolution of Cycle 4 Startup Commitment on Measurement Uncertainity Values, W. Counsil to R. Clark, January 8, 1981.
810401	NNECO letter, Resolution to Cycle 4 Startup Commitment on LOCA Asymmetric Blowdown Loads, W. Counsil to R. Clark, April 1, 1981.
810407	NRC Issues Amendment No. 66, TMI Category A TS, R. Clark to W. Counsil, April 7, 1981.
810608	NNECO letter, Results of LOCA Asymmetric Blowdown Loads for Mixed Core, W. Counsil to R. Clark, June 8, 1981.
810622	NRC letter, Acceptance of Physics Portion of BSR, R. Clark to W. Counsil, June 22, 1981.

Reference No.	Description
810720	Westinghouse letter, Large Break LOCA Results for Millstone-2, E. Rahe to N. Lauben, July 20, 1981.
810917	NNECO letter, CEA Guide Tube Wear, W. Counsil to R. Clark, September 11, 1981.
810928	NNECO letter, Resolution of Cycle 4 Startup Commitment on CEA Guide Tube Wear Evaluation Program, W. Counsil to R. Clark, September 28, 1981.
811015	NNECO letter, Resolution of Cycle 4 Startup Commitment on Cycle 5 Reload Outage Steam Generator Inspection Program, W. Counsil to R. Clark, October 15, 1981.
811016	NNECO letter, Resolution of Cycle 4 Startup Commitment on Worst Large Break LOCA Burnup of Westinghouse Fuel in CE Designed Core, W. Counsil to R. Clark, October 16, 1981.
811117	NNECO letter, Cycle 5 Reload Safety Analysis (RSA), J. Cagnetta to R. Clark, November 17, 1981.
811201	NRC Approval of Westinghouse Appendix K Evaluation Model (Ref. 810515), J. Miller to E. Rahe, December 1, 1981.
811202	NNECO Application for Shutdown Cooling and Coolant Circulation During Refueling, R. Werner to R. Clark, December 2, 1981.
811817	NNECO Application for Cycle 5 Technical Specification Changes, J. Cagnetta to R. Clark, December 17, 1981.
811218	NRC Issues Amedment No. 71, Shutdown Cooling and Coolant Circulation During Refueling, E. Conner to W. Counsil, December 18, 1981.
811224	NRC letter, Request for Additional Information on RSA, C. Trammell to W. Counsil, December 24, 1981.
820108	NNECO letter, Cycle 5 Reload Fuel, W. Counsil to R. Clark, January 8, 1982.
820112	NRC letter, Acceptance of Section 5.3.2 through 5.3.9, 5.3.13 and 5.3.15 through 5.3.17 of the BSR, R. Clark to W. Counsil, January 12, 1982.
820114	NNECO Application for Additional Cycle 5 Technical Specification Change, W. Counsil to R. Clark, January 14, 1982.

Reference No.	Description
820204-1	NNECO letter, Additional Information on Cycle 5 Reload, W. Counsil to R. Clark, February 4, 1982.
820204-2	NNECO letter, Transient and Accident Analyses with Additional Steam Generator Tubes Plugged, W. Counsil to R. Clark, February 4, 1982.
820218	NRC letter, Acceptance of Remaining Sections of BSR, R. Clark to W. Counsil, February 18, 1982.
820219	NNECO letter, Large Break LOCA/ECCS Performance Results with Additional Steam Generator Tubes Plugged, W. Counsil to R. Clark, February 19, 1982.
820222	NRC Issues Amendment No. 72, Cycle 5 Miscellaneous Technical Specification Changes, E. Conner to W. Counsil, February 22, 1982.
820223	NNECO letter, Fission Gas/Clad Collapse Considerations During Cycle 5, W. Counsil to R. Clark, February 23, 1982.
820301	NNECO letter, Supplemental Information on Plugged Steam Generator Tubes, W. Counsil to R. Clark, March 1, 1982.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-336

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 74 to Facility Operating License No. DPR-65, issued to Northeast Nuclear Energy Company, the Connecticut Light and Power Company, the Hartford Electric Light Company, and the Western Massachusetts Electric Company (the licensee), which revised Technical Specifications for operation of the Millstone Nuclear Power Station, Unit No. 2 (the facility) located in the Town of Waterford, Connecticut. The amendment is effective as of the date of issuance.

The amendment changes the Technical Specifications to authorize Cycle 5 operation at 2700 MWt with a mixed core with two-thirds Westinghouse fuel and one-third Combustion Engineering fuel and modified (sleeved, reduced flow and insert) guide tubes for the control element assemblies. The amendment also incorporates changes resulting from the analysis of the Cycle 5 reload with Westinghouse fuel, removes the requirements for mid-cycle moderator temperature coefficient determination, redefines the fully withdrawn position of the regulating control rods, and modifies the pressurizer level operational band.

The applications for the amendment comply 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 applications for amendment dated December 17, 1981 and January 14, 1982, as supplemented, (2) Amendment No. 74 to License No. DPR-65, 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 Waterford Public Library, Rope Ferry Road, Waterford, Connecticut. 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 Licensing.

Dated at Bethesda, Maryland, this 5th day of March, 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
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