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

DEC 3 0 1983

Docket No. 50-336

Mr. W. G. Counsil, Senior Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
P.O. Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Counsil:

The Commission has issued the enclosed Amendment No. 90 to Facility Operating License No. DPR-65 for Millstone Nuclear Power Station, Unit 2, in response to your application dated April 13, 1983, and supplements dated June 2, July 1, August 12, September 15, November 2, 4, and 17, and December 1, 1983. These changes reflect changes to the Technical Specifications necessary for plant operation in Cycle 6. These changes include:

- Reduced Reactor Coolant Flow Rate
- Reduced CEA Drop Time
- New Axial Shape Index Tent
- Revised Total Planar Peaking Factor Curve
- Revised Total Radial Peaking Factor Curve.
- Revised Auxilary Feedwater Pump LCOs and Surveillance Requirements.

The above referenced supplements:

- Provided discussion of fuel degradation and failed fuel assemblies.
- Provided discussion of plant operation with broken fuel holdown springs
- Provided reassessments of LOCA type accidents with thermal shield removed.
- Provided a redesign of Cycle 6 to reflect changes made to the planned fuel loading, and
- Provided, additional information to support Cycle 6 and analysis of STGR's.

In our letter dated November 14, 1983, we transmitted to you a draft Safety Evaluation (SE) concerning the non-LOCA transient and accident analyses for the Cycle 6 reload, and your qualifications to perform licensing calculations. That draft SE is incorporated in this SE. The confirmatory items requested in our letter dated November 14, 1983 remain outstanding.

Additionally, in response to your analysis of LOCA consequences after the removal of the thermal shield, we request you commit to provide by the next refueling outage, either:

- Confirmation that the peak clad temperature does not increase by more than 20°F, or

- An ECCS analysis for the limiting large break (per Section II.1.b. to 10 CFR Part 50, Appendix K).

We request that your commitment to provide this additional confirmatory submittal be made within 60 days of your receipt of this letter. The information requested affects fewer than 10 respondents; therefore OMB clearance is not required under P.L. 96-511.

A copy of our Safety Evaluation is enclosed. The notice of issuance will be included in the Commission's next monthly Federal Register notice.

Sincerely,

Kenneth L. Heitner

Kenneth L. Heitner, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 90 to DPR-65
2. Safety Evaluation

cc: See next page

Northeast Nuclear Energy Company

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

NORTHEAST NUCLEAR ENERGY COMPANY

THE CONNECTICUT LIGHT AND POWER COMPANY

THE WESTERN MASSACHUSETTS ELECTRIC COMPANY

DOCKET NO. 50-336

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 90
License No. DPR-65

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

   A. The application for amendment by Northeast Nuclear Energy Company, et al. (the licensee) dated April 13, 1983, as supplemented, 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 (1) 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. 90 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective on the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 30, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 90

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

Remove and 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 provided to maintain document completeness.

Remove	Insert
2-2	2-2
2-4	2-4
3/4 1-26	3/4 1-26
3/4 2-4	3/4 2-4
3/4 2-6	3/4 2-6
3/4 2-8	3/4 2-8
3/4 2-9	3/4 2-9
3/4 2-14	3/4 2-14
3/4 7-4	3/4 7-4

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and maximum cold leg coolant temperature shall not exceed the limits shown on Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of maximum cold leg temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

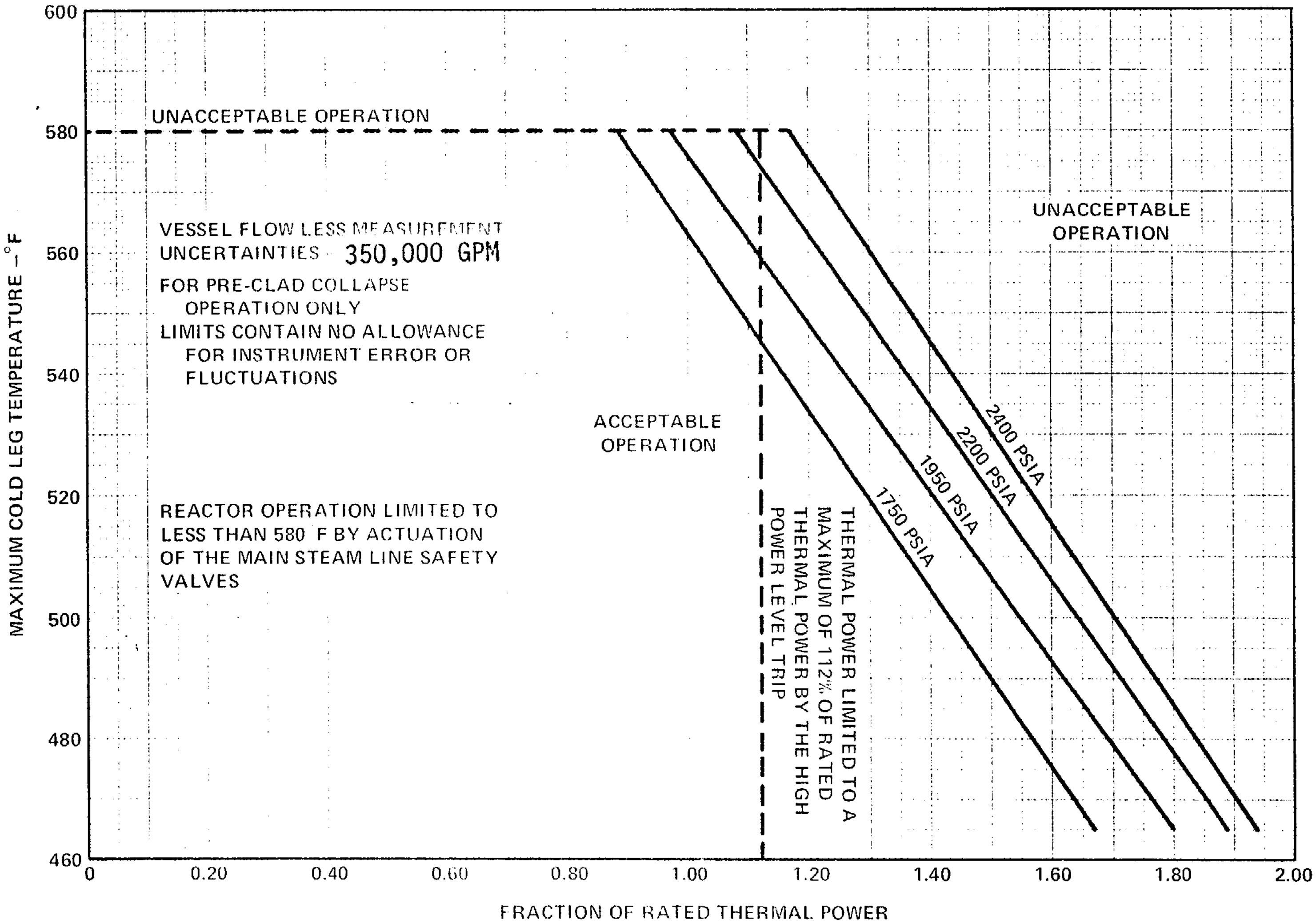

FIGURE 2.1-1 Reactor Core Thermal Margin Safety Limit – Four Reactor Coolant Pumps Operating

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: AS SHOWN FOR EACH CHANNEL IN TABLE 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Level-High		
Four Reactor Coolant Pumps Operating	$\leq$ 9.6% above THERMAL POWER, with a minimum setpoint of $\leq$ 14.6% of RATED THERMAL POWER, and a maximum of $\leq$ 106.6% of RATED THERMAL POWER.	$\leq$ 9.7% above THERMAL POWER, with a minimum of $\leq$ 14.7% of RATED THERMAL POWER, and a maximum of $\leq$ 106.7% of RATED THERMAL POWER.
3. Reactor Coolant Flow - Low (1)		
Four Reactor Coolant Pumps Operating	$\geq$ 91.7% of reactor coolant flow with 4 pumps operating*.	$\geq$ 90.1% of reactor coolant flow with 4 pumps operating*.
4. Reactor Coolant Pump Speed - Low	$\geq$ 830 rpm	$\geq$ 823 rpm
5. Pressurizer Pressure - High	$\leq$ 2400 psia	$\leq$ 2408 psia
6. Containment Pressure - High	$\leq$ 4.75 psig	$\leq$ 5.23 psig
7. Steam Generator Pressure - Low (2) (5)	$\geq$ 500 psia	$\geq$ 492 psia
8. Steam Generator Water Level - Low (5)	$\geq$ 36.0% Water Level - each steam generator	$\geq$ 35.2% Water Level - each steam generator
9. Local Power Density - High (3)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2 (4).	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2 (4).

* Design Reactor Coolant flow with 4 pumps operating is 350,000 gpm.

POSITION INDICATOR CHANNELS (Continued)

LIMITING CONDITION FOR OPERATION

b) The CEA group(s) with the inoperable position indicator is fully inserted, and subsequently maintained fully inserted, while maintaining the withdrawal sequence and THERMAL POWER level required by Specification 3.1.3.6 and when this CEA group reaches its fully inserted position, the "Full In" limit of the CEA with the inoperable position indicator is actuated and verifies this CEA to be fully inserted. Subsequent operation shall be within the limits of Specification 3.1.3.6.

c. With a maximum of one reed switch position indicator channel per group or one pulse counting position indicator channel per group inoperable and the CEA(s) with the inoperable position indicator channel at either its fully inserted position or fully withdrawn position, operation may continue provided:

1. The position of this CEA is verified immediately and at least once per 12 hours thereafter by its "Full In" or "Full Out" limit (as applicable),

2. The fully inserted CEA group(s) containing the inoperable position indicator channel is subsequently maintained fully inserted, and

3. Subsequent operation is within the limits of Specification 3.1.3.6.

d. With one or more pulse counting position indicator channels inoperable, operation in MODES 1 and 2 may continue for up to 24 hours provided all of the reed switch position indicator channels are OPERABLE.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each position indicator channel shall be determined to be OPERABLE by verifying the pulse counting position indicator channels and the reed switch position indicator channels agree within 6 steps at least once per 12 hours except during time intervals when the Deviation circuit is inoperable, then compare the pulse counting position indicator and reed switch position indicator channels at least once per 4 hours.

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) CEA drop time, from a fully withdrawn position, shall be $\leq$ 2.75 seconds from when electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with:

a. $T_{avg} \geq 515°F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODE 3.

ACTION:

a. With the drop time of any full length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full length CEAs shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal of the reactor vessel head,

b. For specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At least once per 18 months.

TABLE 4.7-1

STEAM LINE SAFETY VALVES

VALVE NUMBERS	LIFT SETTING (± 1%)	ORIFICE SIZE
a. 2-MS-246 & 2-MS-247	1000 psia	4.515 $in.^2$
b. 2-MS-242 & 2-MS-254	1005 psia	4.515 $in.^2$
c. 2-MS-245 & 2-MS-249	1015 psia	4.515 $in.^2$
d. 2-MS-241 & 2-MS-252	1025 psia	4.515 $in.^2$
e. 2-MS-244 & 2-MS-251	1035 psia	4.515 $in.^2$
f. 2-MS-240 & 2-MS-250	1045 psia	4.515 $in.^2$
g. 2-MS-239, 2-MS-243, 2-MS-248 & 2-MS-253	1050 psia	4.515 $in.^2$

PLANT SYSTEMS

AUXILIARY FEEDWATER PUMPS

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three steam generator auxiliary feedwater pumps shall be OPERABLE with:

a. Two feedwater pumps capable of being powered from separate OPERABLE emergency busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Starting each pump from the control room,

2. Verifying that:

a) Each motor driven pump develops a discharge pressure of ≥1070 psig on recirculation flow, and

b) The steam turbine driven pump develops a discharge pressure of ≥1080 psig on recirculation flow when the secondary steam supply pressure is greater than 800 psig. The provisions of Specification 4.0.4 are not applicable for entry into Mode 3.

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

SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO ISSUANCE OF AMENDMENT NO. 90 TO DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY, EL AL.

MILLSTONE UNIT 2

DOCKET NO. 50-336

1. Introduction and Background

In Reference 1, Northeast Nuclear Energy Company (NNECO or licensee) submitted a license amendment request and the Reload Safety Analyses (RSA) in support of the Millstone Unit No. 2, Cycle 6 reload. As indicated in the submittal, the bases on which the Cycle 6 reload was analyzed were documented in a "Basic Safety Report" (BSR) (Ref. 2). The BSR, as supplemented by Reference 3, serves as the reference fuel assembly and safety analysis report for the use of Westinghouse fuel at Millstone 2 (a Combustion Engineering plant). Reference 4 documents the NRC staff's review and acceptance of the BSR.

By Reference 5, NNECO informed the staff that due to the elevated levels of radioactive iodine and other fission products identified during Cycle 5 operation, NNECO anticipated the discovery of a number of fuel assemblies with leaking fuel rods during the 1983 refueling outage.

Since that time, NNECO performed fuel sipping identifying 26 fuel assemblies with failed fuel rods. In addition, visual examinations revealed 15 fuel assemblies to have broken holddown springs. Further, structural damage was observed in two assemblies, one of which also had a broken holddown spring. This damage was reported to the staff in Licensee Event Reports 50-336/83-25, 83-25/01-T, 83-26, and 83-26/01-T. Reference 5 provided a detailed discussion of the fuel degradation.

As discussed in Reference 6, NNECO is replacing all leaking fuel assemblies with a combination of new and previously discharged fuel assemblies. These changes have necessitated a revised loading pattern for Cycle 6 operation. In addition, assemblies F37 and F73, which sustained some structural damage, are being replaced.

By Reference 7, NNECO reported damage to the thermal shield support system at Millstone Unit No. 2. The extent of this damage resulted in the need for removal of the thermal shield from the core barrel. Reference 8 provides details of NNECO's thermal shield damage recovery program.

In order to assess the impact of a new loading pattern and the removal of the thermal shield, NNECO has had its fuel vendor reevaluate the Reference 1 Reload Safety Analyses in support of Millstone Unit No. 2 Cycle 6 operation. The results of this review were provided as a supplement to the Reload Safety Analyses (Reference 6).

1.1 General Description of Reactor

The Millstone 2 reactor core is comprised of 217 fuel assemblies. Each fuel assembly has a skeletal structure consisting of five (5) Zircaloy guide thimble tubes, nine (9) Inconel grids, a stainless steel bottom nozzle, and a stainless steel top nozzle. One hundred seventy-six fuel rods are arranged in the grids to form a 14x14 array. The fuel rods consist of slightly enriched uranium dioxide ceramic pellets contained in Zircaloy-4 tubing which is plugged and seal welded at the ends to encapsulate the fuel.

Nominal core design parameters utilized for Cycle 6 are as follows:

Core Power (Mwt)	2,700
System Pressure (psia)	2,250
Reactor Coolant Flow (GPM)	350,000
Core Inlet Temperature (°F)	549
Average Linear Power Density (kw/ft) (based on best estimate hot, densified core average stack height of 136.4 inches)	6.065

The core loading pattern for Cycle 6 consists of twenty-four (24) interior feed assemblies containing 2.7 w/o U-235 and forty-eight (48) peripheral feed assemblies containing 3.2 w/o U-235. These are replacing seventy-two (72) Combustion Engineering (CE) batch E assemblies. Due to fuel defects in Cycle 5 and subsequent symmetry consideration, sixteen (16) interior feed assemblies containing 2.70 w/o U-235, twenty (20) CE assemblies from Batch A and one (1) CE assembly from Batch B (these CE assemblies were discharged at the end of Cycle 1) are needed as well.

2. Fuel System Design

The fuel system design for Millstone Unit 2, Cycle 6 is the same as that approved (Ref. 4) for Cycles 4 and 5. That is, approval of the BSR constituted approval of the use of a mixed core of Combustion Engineering and Westinghouse fabricated fuel assemblies. The replacement of CE fuel with Westinghouse fuel at each reloading would eventually lead to a core with all Westinghouse fuel.

The failed fuel assemblies at Millstone necessitated a revision to the reload plan such that a mixed core, as described in Section 1.1, results. The reload redesign is a result of the following:

1. Fuel rod failures in 26 assemblies
2. Removal of two damaged fuel assemblies
3. Removal of the thermal shield
4. Failure of holddown springs in 15 fuel assemblies.

As described in Reference 9, the reload redesign utilizes a combination of new and previously discharged fuel assemblies to replace the leaking and broken fuel assemblies. Since this redesign uses previously approved fuel assembly types and since the redesign and the reinserted CE assemblies will not receive greater than design exposure, the redesign is acceptable from the fuel system point of view.

The NRC was informed of the broken holddown springs identified on 15 fuel assemblies by Reference 10. A summary of information discussed at a meeting on October 12, 1983 on the broken holddown springs was presented in Reference 6. At this meeting, NNECO documented plans to evaluate the replacement of the broken holddown springs. A repair procedure and tooling was developed to effect the replacement of the holddown springs on irradiated fuel assemblies. This procedure was utilized successfully on one fuel assembly. However, NNECO decided that the irradiated fuel repair procedure involved a high risk with the potential for damaging fuel assemblies, particularly fuel pins, during the repair.

NNECO therefore reached the conclusion and provided supporting analysis (Ref. 11) that operation of Cycle 6 with 9 fuel assemblies, each with a single broken holddown spring, is acceptable and prudent. The analysis provided by NNECO characterizes the breaks to the holddown springs, provides justification that the breaks were caused by excessive vibratory motion during reactor operation, discusses fretting wear, loose parts, control rod jamming and the probability of multiple fractures, and concludes that operation of Cycle 5 with the 9 assemblies having broken holddown springs would be acceptable. This is primarily because the number of active turns of the springs is only slightly decreased by the types of breaks observed. Future new fuel will have newly designed springs.

We have reviewed the material provided by NNECO and agree with the conclusion that operation of Cycle 6 with 9 assemblies containing broken holddown springs will not pose a significant reduction in safety of the power plant.

3.0 Nuclear Design

The nuclear design procedures and models used for the analysis of the Millstone Unit 2 Cycle 6 reload core (Reference 1) are the same as those used for Cycle 5. These are documented in the Millstone Unit 2 Basic Safety Report (BSR) (Reference 2) and have been approved (Reference 4) for the analysis of the Millstone Unit 2 core using Westinghouse reload fuel beginning with Cycle 4.

The licensee provided a tabular summary (Table 2, Reference 1) of the changes in the Cycle 6 kinetics characteristics compared with the current limits based on the most limiting BSR safety analysis and the Cycle 4 and 5 analyses. All of the Cycle 6 values fell within the current limits. The kinetics parameters were, therefore, acceptable for use in the Cycle 6 accident analysis because they are calculated with approved methods, and they are within the bounds of values previously approved.

The reanalysis of the reload performed as a result of the fuel failures (Reference 9) and removal of the thermal shield was performed with the same approved techniques discussed above. In Reference 9, Table 2 the kinetics parameters for the Cycle 6 reload redesign are given. These are all within the current limits with a small exception in the least negative and above 30%

power doppler temperature coefficients and the maximum delayed neutron fraction. The licensee examined the effects of these changes on accident analyses in Reference 9, pages 7 and 8, with the conclusion that the potential effects were small, and no reanalyses were necessary. We reviewed these evaluations and agree that the small changes in these parameters do not lead to a need for reanalyses of any accidents, and that the revised fuel loading and removal of the thermal shield is acceptable with respect to nuclear design.

The control rod worths and shutdown requirements for the Cycle 6 redesign and the initial Cycle 6 design are presented in Table 3 of Reference 9 and compared with previous Cycle 5 values. At EOC 6, the reactivity worth with all control rods inserted assuming the highest worth rod is stuck out of the core is 6.00% assuming a 10% reduction to allow for uncertainty. The reactivity worth required for shutdown, including the contribution required to control the steamline break event at EOC 6 is 5.92% . Therefore, sufficient control rod worth is available to accommodate the reactivity effects of the steamline 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. We have reviewed the calculated control rod worths and the uncertainties in these worths based upon comparison of calculations with experiments presented in the BSR and in previous Westinghouse reports. On the basis of our review, we have concluded that the 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 most reactive control rod is stuck in the fully withdrawn position.

The total trip reactivity as a function of position calculated for Cycle 6 was more limiting than that calculated for Cycle 5. The Cycle 6 curve was therefore used in all accident reanalysis.

4.0 Thermal-Hydraulic Design

Millstone 2 Cycle 6 utilized the Basic Safety Report (Ref. 2) which was approved by the staff in Reference 4. The Basic Safety Report was also used as the basis for Cycles 4 and 5 operation.

As discussed in the BSR, the Westinghouse fuel assemblies have been designed and shown through testing to be hydraulically compatible with all resident Millstone 2 fuel assemblies. A detailed discussion is given in the staff SER of Cycle 4 dated October 6, 1980 (Ref. 12).

The DNB analysis for Cycle 6 was performed for a minimum reactor coolant flow rate of 350,000 gpm and a radial peaking factor, F_r, of 1.565. A reduction in flow from 370,000 gpm to 362,600 gpm and a conservative reduction in F_r from 1.63 to 1.597 was previously implemented during Cycle 5 operation. As indicated by the power and flow sensitivities reported in the Cycle 4 Reload Safety Evaluation Report (Ref. 13) a flow reduction can be offset by a power (or F_r) reduction in a 2:1 ratio to maintain a constant DNBR. Thus the reduction in flow has been more than offset by the reduction in radial peaking factor and this has been confirmed by the licensee in their Cycle 6 analysis. The Cycle 6 analysis takes a partial credit of 3.0% of the net conservatism which exists between convoluting and summing the uncertainties of various measured plant power parameters in terms of power. This partial credit was applied in previous cycles and its approval is discussed in more detail in the Cycle 4 Reload Safety Evaluation Report (Ref. 13); therefore, we find operation of Cycle 6 acceptable.

5.0 Accident Analysis

5.1 CEA Withdrawal at Power

The CEA withdrawal at power accident was reanalyzed for Cycle 6 to assess the impact of increased steam generator tube plugging and the corresponding reduction in flow. The results of this analysis show that the thermal margin low pressure trip maintains the minimum DNBR above 1.30 over the full range of reactivity insertion rates, which is acceptable.

5.2 LOCA Reanalysis with 15.3% Steam Generator Tube Plugging

Millstone Unit 2 is currently licensed to operate at a peak core linear power of 15.6 kw/ft with up to 9.4% of the steam generator tubes plugged (reference 14). In anticipation of further degradation in the steam generators, the licensee has submitted a revised large break and small break LOCA analysis for 15.3% tube plugging (Reference 15).

The limiting large break (C_d=0.6) was calculated with the currently approved Westinghouse evaluation model (EM), and found to conform to the acceptance criteria of 10 CFR 50, Appendix K. The increase from 9.4% plugging to 15.3% plugging produced a 10°F increase in the calculated peak clad temperature (from 2045°F to 2055°F). We asked the licensee to explain this surprisingly small increase in PCT, in view of previous Westinghouse EM calculations showing much higher sensitivity of PCT to tube plugging (reference 3). The licensee responded that Combustion Engineering plants, such as Millstone 2, differ from Westinghouse plants in several important respects. First, the worst case large LOCA for Millstone 2 is with the primary coolant pumps running, and the blowdown is less sensitive to the resistance of the steam generator. Further, the C-E containment pressure is a few PSI higher than for W plants, and the locked rotor resistance is lower. Because of these two facts, the calculated reflood rate never falls below 1 inch/second, and the requirement to use steam cooling is not invoked. The resulting calculated PCT of 2055°F is considerably lower than the value used for the sensitivity study in reference 3. At the lower PCT, there is considerably less zirconium-water reaction, and peak clad temperature is less sensitive to system changes such as tube plugging.

The effect of additional tube plugging (up to 18%) on the limiting small break was also reevaluated and shown to be minimal.

Based on the LOCA analysis, we conclude that operation with 15.3% steam generator tube plugging does not require a reduction in the technical specification limit of 15.6 KW/ft peak linear power.

5.3 LOCA Consequences After Removal of Thermal Shield

By letter dated November 17, 1983, (Reference 17) NNECO submitted a supplement to its Cycle-6 reload application with regard to removal of the thermal shield from the core barrel. The licensee presented qualitative arguments to support the validity of previous LOCA analyses which did not account for the removal of the thermal shield. The following are those arguments.

Removal of the thermal shield increases the downcomer coolant volume by 120 $ft.^3$ For a small break LOCA event, the added inventory is beneficial since additional margin to core uncovery is provided. For those break sizes resulting in core uncovery, the added margin would lead to a lower calculated peak clad temperature. The staff therefore does not require reanalysis of small break LOCAs.

Removal of the thermal shield could result in a somewhat higher calculated peak clad temperature for a large break LOCA. This is attributed, in part, to additional time required to replenish coolant to the downcomer prior to beginning of reflood. The licensee estimates a 2 second delay in the time to bottom of core recovery (BOCREC). Since the calculated peak clad temperature (PCT) prior to removal of the thermal shield is only 2055^oF, there exists ample margin to the 10 CFR 50.46 limit of 2200^oF. We are confident, therefore, that reanalysis of the limiting large break with the thermal shield removed would not result in a calculated PCT in excess of 2200^oF. We therefore find LOCA consequences of removing the thermal shield are acceptable on condition that the licensee confirm, prior to next refueling outage, either that the calculated peak clad temperature does not increase by more than 20^oF or submit an ECCS analysis for the limiting large break (per Section II.1.b to 10 CFR Part 50, Appendix K).

5.4 Steam Line Breaks

The steam line break event analyzed in support of the Cycle-6 reload was calculated by Westinghouse. Based on the licensee's response to NRC question 440.1, it appears that Westinghouse did not model thermal-hydraulic and neutronic asymmetry. A large steam line break will lead to complex asymmetric thermal-hydraulics and neutronics within the reactor vessel. This will result in greater moderator feedback at the core quadrant nearest the affected loop (loop with the broken steam line).

In response to question 440.2, the licensee stated:

> ...when mixing is good, the upper head temperature tends to fall at a slower rate, due to the fact that more flow from the cold loop is allowed to mix with the hot loop flow in the inlet and outlet of the vessel. Part of the inlet flow is routed to cool the upper head. Since this water is warmer than in the poor mixing case, the upper head temperature would be higher.

The licensee assumed homogeneous reactor coolant mixing in the upper head of the reactor vessel when analyzing the steam line break event. After the pressurizer is emptied of liquid inventory, the primary system pressure is governed by the saturation temperature in the upper head. Neglecting a separate model for the upper head and its associated metal wall heat capacity, the calculated system depressurization could be non-conservatively low (by 300-400psi), thereby resulting in excessive ECCS injected boron. The lack of upper head mixing was observed in the St. Lucie-1 natural circulation cooldown event and resulted in the NSSS vendor (C-E) changing its analytical model. The Westinghouse methodology did not account for this in the steam line break analysis. We require justification that the analytical model used adequately addresses this phenomenon.

The licensee assumed complete mixing of the fluid from the intact and affected coolant loops as it enters the reactor vessel. Both Combustion Engineering and Westinghouse have developed proprietary data to credit some mixing, which is advantageous for DNBR consideration. Assuming ideal or complete mixing requires additional justification.

The licensee analyzed the limiting steam line break to occur for zero power conditions with offsite power available. Qualitative discussions were presented in response to NRC question 440.1 to address steam line breaks with loss of offsite power. The licensee stated:

> Since the reactor coolant pumps are coasting down with the loss of offsite power, the ability of the emptying steam generator to extract heat from the reactor coolant system is reduced. The closest approach to criticality would occur later in the transient and the core power increase would be slower than in the similar case with offsite power available.

The staff agrees that during natural circulation the primary system will depressurize less. However, it is not obvious that the event with loss of offsite power would not be more severe, when accounting for asymmetric thermal-hydraulics. We base our concern on previous vendor and staff calculations which showed 50°F to 100°F lower coolant temperatures in the affected loop during pump coastdown versus the case with offsite power available. Consequently, the moderator reactivity feedback could be significantly different, with the pump coastdown event being more severe. Since Westinghouse specifically designed a mitigating system for its plants which initiates a safety injection actuation signal during a steam line break event, the above concerns may not be applicable to Westinghouse plants. However, Millstone 2 was designed by Combustion Engineering, does not have the Westinghouse protective system, and consequently has its ECCS initiated much later into the transient (relative to a Westinghouse plant). Consequently, the case with loss of offsite power for Millstone 2 has a potential for being limiting.

We request further justification to demonstrate the acceptability of the steam line break analysis performed by Westinghouse. In addition, we request confirmation that General Design Criterion 17 (GDC-17) is met for the steam line break event. This requires an assessment of the steam line break event with loss of offsite power.

5.5 CEA Ejection and Seized Rotor Events

The licensee's evaluation of the CEA Ejection and Seized Rotor events is not entirely acceptable. The events were assessed on a peak clad temperature (PCT) criterion. Unless well founded and clearly acceptable fuel failure criteria based on a PCT and pellet-clad-interaction (PCI) are presented, all rods experiencing a DNBR below the 95/95 limit should be assumed to fail. This criterion has been a long standing NRC position. We therefore request a confirmatory evaluation to demonstrate that the offsite radiological consequences are acceptable assuming failure of all fuel rods that have a calculated DNBR below the 95/95 DNBR limit.

5.6 Steam Generator Tube Rupture

The steam generator tube rupture event was analyzed by the licensee using the RETRAN computer program. In support of the RETRAN nodalization for Millstone-2, the licensee submitted a calculation of a turbine trip event and compared it with plant data. The calculated parameters of interest (i.e., pressure, temperature and pressurizer level) were in good agreement with the data. However, the transient was mild and did not challenge the model such that its applicability to more severe events is established. We therefore require the licensee to demonstrate the applicability of the model to a steam generator tube rupture event.

Specifically, the nodalization of the upper head may be inappropriate for events leading to voiding within that region. Thermal-hydraulic behavior in the upper head could significantly alter the consequences of such an event. Ideal mixing of fluid in the upper head with the upper plenum coolant may be inappropriate. Similarly, lumping the steam generator inlet plenum with half of the steam generator tubes could result in improper primary system thermal conditions. Staff evaluations with similar codes have show that finer nodalization is reouired for modeling such events.

The initial conditions for the steam generator tube rupture analysis consisted of nominal operating conditions and instrumentation uncertainties. This initialization may be appropriate for a most probable consequences assessment but may not be appropriate for licensing evaluations. Licensing calculations should bound the operating conditions of the plant. Typically, these are determined by the technical specifications for plant operation. The licensee should perform the analysis at the technical specification limits or otherwise show that the conditions at which the plant was analyzed provide bounding results for all allowed plant operating conditions.

The licensee has not incorporated the limiting single failure in its steam generator tube rupture analysis. The consequences of the limiting active failure with and without offsite power available must be assessed.

6.0 Open Items Requiring Confirmatory Response

We have reviewed the licensee's submittal of the Cycle-6 reload for Millstone 2 and, to a limited extent, the qualifications of NNECO to perform licensing analyses. We require the licensee to demonstrate that the analyses submitted in support of the Cycle-6 reload were conservative and comply with regulatory requirements and criteria. In addition, the licensee must demonstrate qualification for performing licensing analyses and demonstrate understanding of plant responses under transient and accident conditions, as requested in NRC Generic Letter 83-11.

A copy of the Millstone-2 RETRAN deck for the steam generator tube rupture analysis should be submitted to the NRC. The staff will examine the input and modeling techniques as part of the qualification review of the licensee.

The licensee has reanalyzed the steam line break, the CEA ejection, the seized rotor and the steam generator tube rupture events in support for the Cycle-6 reload. However, the licensee did not evaluate the consequences resulting from a postulated loss of offsite power for the above events, as required by GDC-17 nor has the licensee postulated the limiting active single failure in accordance with present and past regulatory practices.

The staff is unable to conclude on the acceptability of the submitted analyses unless the licensee either:

(a) Reanalyzes the above events assuming the limiting single failure, with and without a loss of offsite power (as required by GDC-17) while using an acceptable model, or
(b) Provides justification and/or requests an exemption for deviating from current regulatory requirements.

Conditional upon the licensee's commitment to acceptably respond to our concerns listed above, we believe that the confirmatory analyses would not substantially alter the conclusions for these events and the continued operation of the plant does not endanger the health and safety of the public.

Our acceptance of the confirmatory submittal will be conditional upon an acceptable inspection of the licensee's quality assurance (QA) program as applied to computer code development and use practices. This inspection will be performed by NRC Region IV during the first week of February, 1984. In addition, NRR will further assess the qualifications of Northeast Utilities to perform licensing submittals with the RETRAN computer program. Northeast Utilities should provide a detailed presentation, at the time of the QA inspection, of their analytical qualifications and understanding of plant responses to postulated transient and accident events. We also request that during that meeting, the licensee submit a copy of their RETRAN deck for the steam generator tube rupture calculation.

7.0 Technical Specification Changes

Technical Specification changes proposed by the licensee in Reference 1 are acceptable as follows. No additional Technical Specification changes were required as a result of the reload reanalysis.

A. Reduced Reactor Coolant Flow Rate - This proposed change affects pp. 2-2, 2-4, and 3/4 2-14 of the Technical Specifications. It involves lowering the required primary coolant flow rate from 362,500 gpm to 350,000 gpm. This new lower flow is established to correspond to a plugging level of 2500 steam generator tubes, and was used in the Cycle 6 analysis. We find it acceptable since it was offset by the reduction in F_r.

B. CEA Drop Time - This proposed change to p. 3/4 1-26 of the Technical Specifications involves a revision of the CEA drop time. At the beginning of Cycle 3, four small flow hole test assemblies were put into the core under CEA locations in an effort to mitigate the guide tube wear problem. At that time, the CEA drop time was changed from 2.75 seconds to 3.1 seconds due to a larger dashpot effect realized with the reduced flow holes. This design is no longer being used as the "guide tube wear" fix at Millstone Unit 2 and the four test assemblies will be removed from the core during this 1983 refueling. The licensee, therefore, proposed changing the CEA drop time back to the original value.

C. New Axial Shape Index Tent - The change to p. 3/42-4 involves a new axial shape index (ASI) monitoring tent for figure 3.2-2 of the Technical Specifications. This tent is used to verify the kw/ft limit of 15.6 which is input to the LOCA analyses. Operation within the tent ensures that the maximum local power is less than 15.6 kw/ft. and thus satisfies the Technical Specification surveillance requirement. Under normal conditions the kw/ft surveillance limit is verified with the incore monitoring system and the only time the ASI tent is used is if the incore system is inoperable.

D. Revised total planar peaking factor, F_{xy}, curve - This change affects pp. 3/4 2-6 and 3/4 2-8 of the Technical Specifications and involves restoring the planar radial peaking factor, F_{xy}, monitoring limits back to the original Beginning of Cycle (BOC) 5 values. The Cycle 6 licensing analyses support this proposed revision.

E. Revised total radial peaking factor (F_r) curve - This proposed change affects pp. 3/4 2-8 and 3/4 2-9 of the Technical Specifications. In comparing the BOC 5 values to BOC 6 values, the required primary flow is being reduced by 5.4% (370,000 gpm to 350,000 gpm). Although the current licensed primary coolant flow rate is 362,600 gpm, BOC 5 values are being used since these values correspond with those of the last transient analysis. The Cycle 4 Reload Safety Analyses have shown that the DNB analysis penalty which results from a reduction of 2% in primary flow can be offset with an approximate 1% reduction in F_r. Therefore, the 4% reduction in allowable F_r more than offsets the penalty associated with a 5.4% reduction in primary flow. The Cycle 6 licensing analyses support this proposed revision.

F. Auxiliary Feedwater Pumps - These proposed changes make Millstone Unit 2 Technical Specifications, specifically p. 3/4 7-4, consistent with NUREG-212, Revision 2 Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors. In addition, the proposed revision modifies the Technical Specifications to reflect the actual plant conditions applicable to Mode 4 under which there is insufficient steam to allow the steam turbine driven auxiliary feedwater pump to meet the required discharge pressure.

These changes are all acceptable because they are consistent with the Cycle 6 licensing analysis, or, in the case of the latter item, make the Millstone Unit 2 Technical Specifications consistent with the accepted specifications of NUREG-212.

8.0 Conclusions

We have reviewed Millstone Unit 2 Cycle 6 reload and the proposed changes to the Technical Specifications and find they are acceptable. The reload uses approved fuel types and will not cause any change in the types or increase in the amount of effluents or any change in the authorized power level of the facility. The transients and accidents, and provisions for reactivity control meet applicable criteria.

8.1 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.

8.2 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: December 30, 1983

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