October 12, 1990

Docket No. 50-336

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Mr. Edward J. Mroczka
Senior Vice President
Nuclear Engineering and Operations
Connecticut Yankee Atomic Power Company
Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Mroczka:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 77063)

The Commission has issued the enclosed Amendment No.148 to Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Unit No. 2, in response to your application dated June 26, 1990, as amended by letter dated August 1, 1990.

The amendment modifies Technical Specifications that have cycle-specific paramenter limits by replacing the values of those limits with a reference to a Core Operating Limits Report (COLR) for the values of those limits. The changes also include the addition of the COLR to the Definitions section and to the reporting requirements of the Administrative Controls section of the Technical Specifications.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

This is also to confirm the verbal commitment of your staff to revise, at a later date, TS Section 6.9.1.7, CORE OPERATING LIMITS REPORT, such that the referenced documents that identify the analytical methods used to determine the core operating limits will be identified by their specific NRC approved revisions and dates of issuance. This will provide a documentation trail that is more direct.

Sincerely,

original signed by

Guy S. Vissing, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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Enclosures:
1. Amendment No. 148
2. Safety Evaluation

cc w/enclosures:
See next page

Document Name: AMEND 77063 M

OFC	:PDI-4:LA	:PDI-4:PM	:PDI-4:D	:OGC	:
NAME	:SNorris	:GVissing/Bah	:JStolz	:[illegible]	:
DATE	:10/1/90	:10/1/90	:10/12/90	:10/4/90	:

DF

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

October 12, 1990

Docket No. 50-336

Mr. Edward J. Mroczka
Senior Vice President
Nuclear Engineering and Operations
Connecticut Yankee Atomic Power Company
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Guy S. Vissing

Guy S. Vissing, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 148
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Edward J. Mroczka
Northeast Nuclear Energy Company

Millstone Nuclear Power Station
Unit No. 2

cc:

Gerald Garfield, Esquire
Day, Berry and Howard
Counselors at Law
City Place
Hartford, Connecticut 06103-3499

R. M. Kacich, Manager
Generation Facilities Licensing
Northeast Utilities Service Company
Post Office Box 270
Hartford, Connecticut 06141-0270

W. D. Romberg, Vice President
Nuclear Operations
Northeast Utilities Service Company
Post Office Box 270
Hartford, Connecticut 06141-0270

D. O. Nordquist
Director of Quality Services
Northeast Utilities Service Company
Post Office Box 270
Hartford, Connecticut 06141-0270

Kevin McCarthy, Director
Radiation Control Unit
Department of Environmental Protection
State Office Building
Hartford, Connecticut 06106

Regional Administrator
Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Bradford S. Chase, Under Secretary
Energy Division
Office of Policy and Management
80 Washington Street
Hartford, Connecticut 06106

First Selectmen
Town of Waterford
Hall of Records
200 Boston Post Road
Waterford, Connecticut 06385

S. E. Scace, Nuclear Station Director
Millstone Nuclear Power Station
Northeast Nuclear Energy Company
Post Office Box 128
Waterford, Connecticut 06385

W. J. Raymond, Resident Inspector
Millstone Nuclear Power Station
c/o U. S. Nuclear Regulatory Commission
Post Office Box 811
Niantic, Connecticut 06357

J. S. Keenan, Nuclear Unit Director
Millstone Unit No. 2
Northeast Nuclear Energy Company
Post Office Box 128
Waterford. Connecticut 06385

Charles Brinkman, Manager
Washington Nuclear Operations
C-E Power Systems
Combustion Engineering, Inc.
12300 Twinbrook Pkwy
Suite 330
Rockville, Maryland 20852

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. 148
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 June 26, 1990, as amended by letter dated August 1, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   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.

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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 Appendix A, as revised through Amendment No. 148 , 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 issuance, to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 12, 1990

ATTACHMENT OF LICENSE AMENDMENT NO. 148

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 areas of change.

Remove	Insert
I	I
XVIII	XVIII
1-5	1-5
2-2	2-2
2-4	2-4
3/4 1-1	3/4 1-1
3/4 1-3	3/4 1-3
3/4 1-5	3/4 1-5
3/4 1-21	3/4 1-21
3/4 1-22	3/4 1-22
3/4 1-28	3/4 1-28
3/4 1-29	3/4 1/29
3/4 1-30	3/4 1-30
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-3a	3/4 2-3a
3/4 2-4	3/4 2-4
3/4 2-9	3/4 2-9
3/4 2-13	3/4 2-13
3/4 2-14	3/4 2-14
3/4 2-15	3/4 2-15
B 3/4 1-1	B 3/4 1-1
B 3/4 1-2	B 3/4 1-2
B 3/4 1-3	B 3/4 1-3
B 3/4 2-1	B 3/4 2-1
6-19	6-19
	6-19(a)
	6-19(b)

DEFINITIONS

INDEX

ADMINISTRATIVE CNTROLS

AXIAL SHAPE INDEX

1.23 The AXIAL SHAPE INDEX (Y_E) used for normal control and indication is the power level detected by the lower excore nuclear instrument detectors (L) less the power level detected by the upper excore nuclear instrument detectors (U) divided by the sum of these power levels. The AXIAL SHAPE INDEX (Y_I) used for the trip and pretrip signals in the reactor protection system is the above value (Y_E) modified by an appropriate multiplier (A) and a constant (B) to determine the true core axial power distribution for that channel.

$$Y_E = \frac{L-U}{L+U} \qquad Y_I = AY_E + B$$

CORE OPERATING LIMITS REPORT

1.24 The CORE OPERATING LIMITS REPORT is the unit specific document that provides the core operating limits for the current operating reload cycle. These cycle specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.7. Plant operation within these operating limits is addressed in individual specifications.

ENCLOSURE BUILDING INTEGRITY

1.25 ENCLOSURE BUILDING INTEGRITY shall exist when:

1.25.1 Each door in each access opening is closed except when the access opening is being used for normal transit entry and exit, and

1.25.2 The enclosure building filtration system is OPERABLE.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

ENGINEERING SAFETY FEATURE RESPONSE TIME

1.27 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of

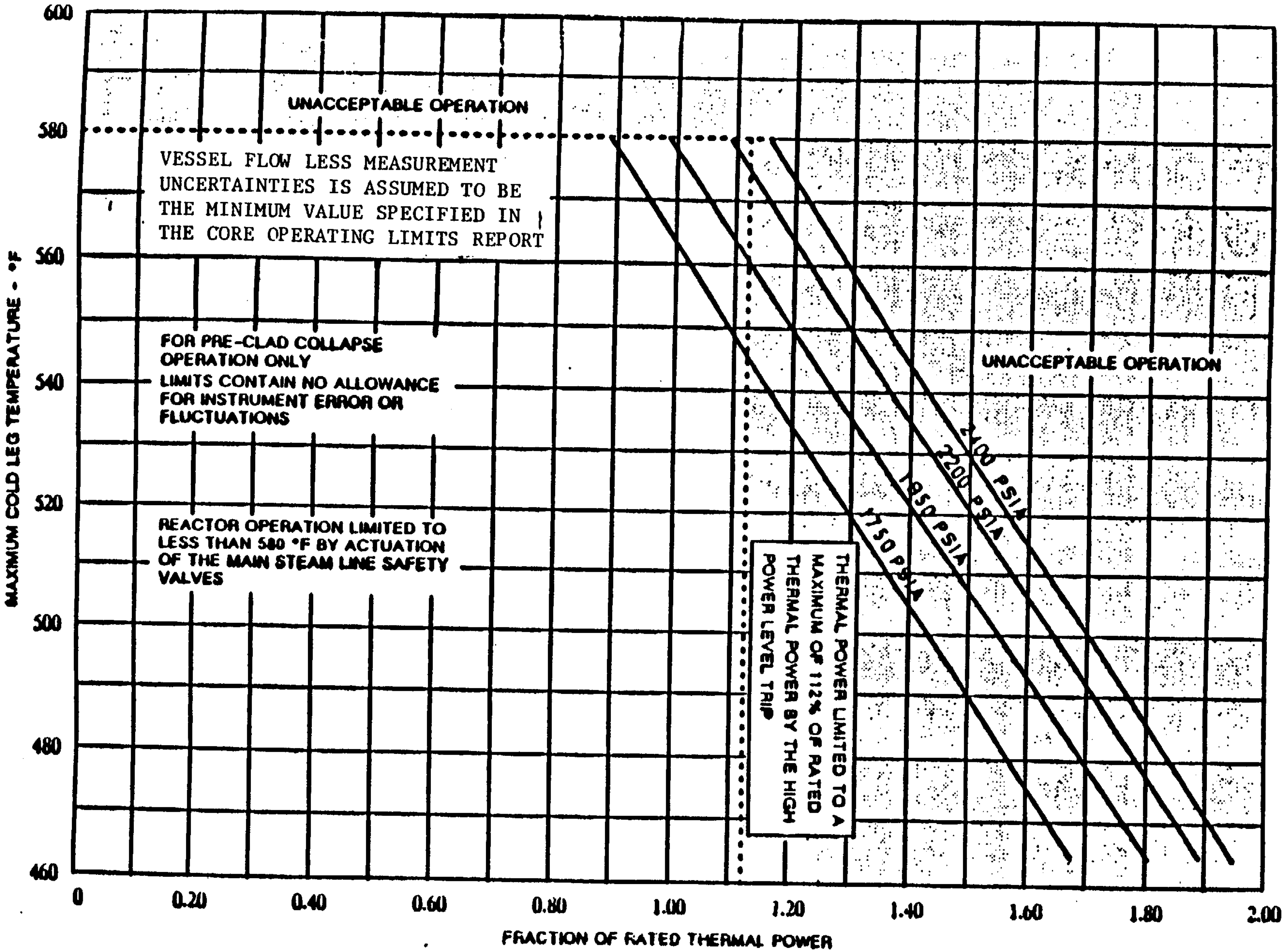

FIGURE 2.1-1 REACTOR CORE THERMAL MARGIN SAFETY LIMIT - FOUR REACTOR COOLANT PUMPS OPERATING

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	≤ 9.6% above THERMAL POWER, with a minimum setpoint of ≤ 14.6% of RATED THERMAL POWER.	≤ 9.7% Above THERMAL POWER, with a minimum of ≤ 14.7% of RATED THERMAL POWER, and a maximum of ≤ 106.7% of RATED THERMAL POWER.
3. Reactor Coolant Flow - Low (1)	≥ 91.7% of reactor coolant flow with 4 pumps operating*.	≥ 90.1% of reactor coolant with 4 pumps operating.
4. Reactor Coolant Pump Speed - Low	≥ 830 rpm	≥ 823 rpm
5. Pressurizer Pressure - High	≤ 2400 psia	≤ 2408 psia
6. Containment Pressure - High	≤ 4.75 psig	≤ 5.24 psig
7. Steam Generator Pressure - Low (2) (5)	≥ 680 psia	≥ 672 psia
8. Steam Generator Water Level - Low (5)	≥ 36.0% Water Level - each steam generator	≥ 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 the lesser of either:
a. The reactor coolant flow rate measured per specification 4.2.6.1, or
b. The minimum value specified in the CORE OPERATING LIMITS REPORT.

3/4.1 REACTVITY CONTROL SYSTEMS

3/4.1.1 REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be within the limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODES 1, 2*, 3 and 4

ACTION:

With the SHUTDOWN MARGIN outside the limit specified in the CORE OPERATING LIMITS REPORT, within 15 minutes initiate and continue boration at ≥ 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 reached.

SURVEILLANCE REQUIREMENT

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be within the limit specified in the CORE OPERATING LIMITS REPORT:

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

SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be within the limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN outside the limit specified in the CORE OPERATING LIMITS REPORT, within 15 minutes initiate and continue boration at ≥ 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

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be within the limit specified in the CORE OPERATING LIMITS REPORT:

a. Immediately upon detection of an inoperable CEA. If the inoperable CEA is immovable or untrippable, the SHUTDOWN MARGIN required by Specification 3.1.1.2 shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA.

b. 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.

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MODERATOR TEMPERATURE COEFFICIENT (MTC)

LIMITING CONDITION FOR OPERATION (Continued)

3.1.1.4 The moderator temperature coefficient (MTC) shall be within the limits specified in the CORE OPERATING LIMITS REPORT. The upper limit shall be less than or equal to:

a. 0.7×10^{-4} ΔK/K/°F whenever THERMAL POWER is ≤ 70% of RATED THERMAL POWER,

b. 0.4×10^{-4} ΔK/K/°F whenever THERMAL POWER IS > 70% of 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 REQUIREMENT

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 comparision with the predicted values.

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

#See Special Test Exemption 3.10.2.

ACTION (Continued):

1. Restored to OPERABLE status within its above specified alignment requirements, or

2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue for up to 7 days per occurrent with a total accumulated time of ≤ 14 days per calendar year provided all of the following conditions are met:

   a.) The THERMAL POWER level shall be reduced to ≤ 70% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination within one hour; if negative reactivity insertion is required to reduce THERMAL POWER, boration shall be used.

   b.) Within one hour after reducing the THERMAL POWER as required by a), above, the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 10 steps of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits of Specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

   c.) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

e. With one full length CEA misaligned from any other CEA in its group by 20 steps or more, reduce THERMAL POWER to ≤ 70% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination within one hour; if negative reactivity insertion is required to reduce THERMAL POWER, boration shall be used. Within one hour after reducing THERMAL POWER as required above, either:

   1. Restore the CEA to within the above specified alignment requirements, or

   2. Declare the CEA inoperable and determine that the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, POWER OPERATION may continue for up to 7 days per occurrence with a total ccumulated time of ≤ 14 days per calendar year provided the remainder of the CEAs in the group with the inoperable CEA are aligned to within 10 steps of the inoperable CEA

ACTION (Continued):

while maintaining the allowable CEA sequence and insertion limits of Specification 3.1.3.6 and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

f. With more than one full length CEA inoperable or misaligned from any other CEA in its group by 20 steps (indicated position) or more, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length CEA shall be determined to be within 10 steps (indicated position) of all other CEAs in its group at least once per 12 hours except during time intervals when the Deviation Circuit and/or CEA Motion Inhibit are inoperable, then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each full length CEA not fully inserted shall be determined to be OPERABLE by movement of at least 10 steps at least once per 31 days.

4.1.3.1.3 The CEA Motion Inhibit shall be demonstrated OPERABLE at least once per 31 days by a functional test of the CEA group deviation circuit which verifies that the circuit prevents any CEA from being misaligned from all other CEAs in its group by more than 10 steps (indicated position).

4.1.3.1.4 The CEA Motion Inhibit shall be demonstrated OPERABLE by a functional test which verifies that the circuit maintains the CEA group overlap and sequencing requirements of Specification 3.1.3.6 and that the circuit prevents the regulating CEAs from being inserted beyond the Transient Insertion Limits of Specification 3.1.3.6:

a. Prior to each entry into MODE 2 from MODE 3, except that such verification need not be performed more often than once per 31 days, and

b. At least once per 6 months.

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3.1.3.6 The regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits specified in the CORE OPERATING LIMITS REPORT. Regulating CEAs are considered to be fully withdrawn when withdrawn to at least 176 steps. CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits is restricted to the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

a. With the regulating CEA groups inserted beyond the Transient Insertion Limits specified in the CORE OPERATING LIMITS REPORT, 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 specified in the CORE OPERATING LIMITS REPORT 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 specified in the CORE OPERATING LIMITS REPORT are not exceeded, or

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

*See Special Test Exception 3.10.2 and 3.10.5.
#With $K_{eff} \geq 1.0$.

LIMITING CONDITION FOR OPERATION (Continued)

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits specified in the CORE OPERATING LIMITS REPORT for intervals > 5 EFPD per 30 EFPD interval or > 14 EFPD per calendar year, except during operations pursuant to the provisions of ACTION items c. and d. of Specification 3.1.3.1, either:

   1. Restore the regulating groups to within the Long Term Steady State Insertion Limits specified in the CORE OPERATING LIMITS REPORT within two hours, or

   2. Be in HOT STANDBY within 4 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Transient Insertion Limits specified in the CORE OPERATING LIMITS REPORT at least once per 12 hours except during time intervals when the PDIL Auctioneer Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulatory CEA groups are inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits specified in the CORE OPERATING LIMITS REPORT shall be determined at least once per 24 hours.

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LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION (Continued)

3.2.1 The linear heat rate, including heat generated in the fuel, clad and moderator, shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1.

ACTION:

During operation with the linear heat rate being monitored by the Incore Detector Monitoring System, comply with the following ACTION:

With the linear heat rate exceeding the limit as indicated by four or more coincident incore channels, within 15 minutes initiate corrective action to reduce the linear heat rate to less than or equal to the limit and either:

a. Restore the linear heat rate to less than or equal to the limit within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

During operation with the linear heat rate being monitored by the Excore Detector Monitoring System, comply with the following ACTIONS:

With the linear heat rate exceeding its limit, as indicated by the AXIAL SHAPE INDEX being outside of the power dependent limits on the Power Ratio Recorder, either:

a. Restore the AXIAL SHAPE INDEX to within the limits specified in the CORE OPERATING LIMITS REPORT within 1 hour from initially exceeding the linear heat rate limit, or

b. Be in at least HOT STANDBY within the next 4 hours.

SURVEILLANCE REQUIREMENT

4.2.1.1 The linear heat rate shall be determined to be within its limits by continuously monitoring the core power distribution with either the excore detector monitoring system or with the incore detector monitoring system.

SURVEILLANCE REQUIREMENT (Continued)

4.2.1.2 Excore Detector Monitoring System - The excore detector monitoring system may be used for monitoring the core power distribution by:

a. Verifying at least once per 12 hours that the full length CEAs are withdrawn to and maintained at or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6.

b. Verifying at least once per 31 days that the AXIAL SHAPE INDEX alarm setpoints are adjusted to within the allowable limits specified in the CORE OPERATING LIMITS REPORT.

4.2.1.3 Incore Detector Monitoring System - The incore detector monitoring system may be used for monitoring the core power distribution by verifying that the incore detector Local Power Density alarms:

a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated at least once per 31 days.

b. Have their alarm setpoint adjusted to less than or equal to the limit when the factors specified in the CORE OPERATING LIMITS REPORT are appropriately included in the setting of these alarms.

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TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F_r^T defined as $F_r^T = F_r (1+T_q)$, shall be within the limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1*.

ACTION:

With F_r^T exceeding its limit within 6 hours either:

a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T to within the limit and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or
b. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENT

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 F_r^T shall be calculated by the expression $F_r^T = F_r (1+T_q)$ and F_r^T shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,
b. At least once per 31 days of accumulated operation in Mode 1, and
c. Within four hours if the AXIMUTHAL POWER TILT (T_q) is > 0.020.

4.2.3.3 F_r shall be determined each time a calculation of F_r^T is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump Combination.

4.2.3.4 T_q shall be determined each time a calculation of F_r^T is required and the value of T_q used to determine F_r^T shall be the measured value of T_q.

*See Special Test Exception 3.10.2

POWER DISTRIBUTION LIMITS

DNB MARGIN

LIMITING CONDITION FOR OPERATION

3.2.6 The DNB margin shall be preserved by maintaining the cold leg temperature, pressurizer pressure, reactor coolant flow rate, and AXIAL SHAPE INDEX within the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its specified limits, restore the parameter to within its above specified limits within 2 hours or reduce THERMAL POWER to ≤ 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.6.1 The cold leg temperature, pressurizer pressure, and AXIAL SHAPE INDEX shall be determined to be within the limits specified in the CORE OPERATING LIMITS REPORT at least once per 12 hours. The reactor coolant flow rate shall be determined to be within the limit specified in the CORE OPERATING LIMITS REPORT at least once per 31 days.

4.2.6.2 The provisions of Specification 4.0.4 are not applicable.

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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, the minimum SHUTDOWN MARGIN specified in the CORE OPERATING LIMITS REPORT 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 the reduced SHUTDOWN MARGIN specified in the CORE OPERATING LIMITS REPORT 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, and 5) 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 flowpaths are provided to ensure single functional capability in the event an assumed failure of a pump or valve renders one of the flowpaths inoperable. Redundant flow paths from the Boric Acid Storage Tanks are achieved through Boric Acid Pumps, gravity feed lines and Charging Pumps. Redundant flow paths from the Refueling Water Storage Tank are achieved through Charging Pump flow path guaranteed by Technical Specification 3.1.2.2 and the HPSI flow path guaranteed by Technical Specification 3.5.2 and 3.5.3. 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 minimum boration capability is sufficient to provide a SHUTDOWN MARGIN within the limits specified in the CORE OPERATING LIMITS REPORT at all temperatures above 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires an equivalent of 4900 gallons of 3.5% boric acid solution from the boric acid tanks plus 15,000 of 1720 ppm borated water from the refueling water storage tank. The refueling water storage tank can also be used alone by feed-and-bleed using well under the 370,000 gallons of 1720 ppm borated water required.

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.

3/4.1.2 BORATION SYSTEMS (Continued)

The analysis to determine the boration requirements assumed that the Reactor Coolant System is borated concurrently with cooldown. In the limiting situation when letdown is not available, the cooldown is assumed to be initiated within 26 hours and cooldown to 220°F, is completed in the next 28 hours.

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.

The boron capability required below 200°F is based upon providing a SHUTDOWN MARGIN within the limit specified in the CORE OPERATING LIMITS REPORT at 140°F after xenon decay. This condition requires either 3750 gallons of 2.5% boric acid solution from the boric acid tanks or 57,300 gallons of 1720 ppm borated water from the refueling water storage tank.

The maximum boron concentration requirement (3.5%) and the minimum temperature requirement (55°F) for the Boric Acid Storage Tank ensures that boron does not precipitate in the Boric Acid System. The daily surveillance requirement provides sufficient assurance that the temperature of the tank will be maintained higher than 55°F at all times.

A minimum boron concentration of 1720 ppm is required in the RWST at all times in order to satisfy safety analysis assumptions for boron dilution incidents and other transients using the RWST as a borated water source as well as the analysis assumption to determine the boration requirement to ensure adequate shutdown margin.

3/4.1.3 MOVEABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of a CEA ejection accident are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met.

The ACTION statements applicable to an immovable or untrippable CEA and to a large misalignment (≥ 20 steps) of two or more CEAs, require a prompt shutdown of the reactor since either

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the AXIAL SHAPE INDEX with two OPERABLE excore neutron flux detectors and verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits specified in the CORE OPERATING LIMITS REPORT using the Power Ratio Recorder. The power dependent limits of the Power Ratio Recorder are less than or equal to the limits specified in the CORE OPERATING LIMITS REPORT. In conjunction with the use of the excore monitoring system and in establishing the AXIAL SHAPE INDEX limits, the following assumptions are made: 1) the CEA insertion limits of Specifications 3.1.3.2, 3.1.3.5 and 3.1.3.6 are satisfied, 2) the flux peaking augmentation factors are as specified in the CORE OPERATING LIMITS REPORT, 3) the AZIMUTHAL POWER TILT restrictions of Specification 3.2.4 are satisfied, and 4) the Total Integrated Radial Peaking Factor does not exceed the limits of Specification 3.2.3.

The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be maintained within the allowable limits specified in the CORE OPERATING LIMITS REPORT. The setpoints for these alarms include allowances, set in the conservative directions as specified in the CORE OPERATING LIMITS REPORT.

3/4.2.3 and 3/4.2.4 TOTAL INTEGRATED RADIAL PEAKING FACTORS F_r^T AND AZIMUTHAL POWER TILT - T_q

The limitations on F_r^T and T_q are provided to 1) ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local power Density - High LCOs and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits, and, 2) ensure that the assumptions used in the analysis establishing the DNB Margin LCO, and Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F_r^T or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the ACTION statements since these additional restrictions provide adequate provisions to assure that the assumptions used in establishing the Linear Heat Rate, Thermal Margin/Low Pressure and Local Power Density - High LCOs and LSSS

d. Documentation of all failures (inability to lift or reclose within the tolerances allowed by the design basis) and challenges to the pressurizer PORVs or safety valves.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, one copy to the Regional Administrator, Region I, and one copy to the NRC Resident Inspector, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.7 a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle.

3/4.1.1.1	SHUTDOWN MARGIN - $T_{avg} > 200°F$
3/4.1.1.2	SHUTDOWN MARGIN - $T_{avg} \leq 200°F$
3/4.1.1.4	Moderator Temperature Coefficient
3/4.1.3.6	Regulating CEA Insertion Limits
3/4.2.1	Linear Heat Rate
3/4.2.3	Total Integrated Radial Peaking Factor - F_r^T
3/4.2.6	DNB Margin

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1) XN-75-27(A), latest Revisions and Supplements, "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," Exxon Nuclear Company.

2) XN-NF-84-73(P), latest Revision and Supplements, "Exxon Nuclear Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Exxon Nuclear Company.

3) XN-NF-82-21(A), latest Revision and Supplements, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company.

4) XN-84-93(A), latest Revision and Supplements, "Steamline Break Methodology for PWR's, Exxon Nuclear Company.

5) XN-75-32(A), Supplements 1, 2, 3, and 4, "Computational Procedure for Evaluation Rod Bow," Exxon Nuclear Company.

6) XN-NF-82-49(A), latest Revision, "Exxon Nuclear Company Evaluation Model EXEM PWR Small Break Model," Exxon Nuclear Company.

7) EXEM PWR Large Break LOCA Evaluation Model as defined by:

XN-NF-82-20(A), latest Revision and Supplements, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," Exxon Nuclear Company.

XN-NF-82-07(A), latest Revision, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company.

XN-NF-81-58(A), latest Revision, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company."

XN-NF-85-16(A), Volume 1 and Supplements, Volume 2 latest Revision and Supplements, "PWR 17x17 Fuel Cooling Test Program," Exxon Nuclear Company.

XN-NF-85-105(A), and Supplements, "Scaling of FCTF Based Reflood Heat Transfer Correlation for Other Bundle Designs," Exxon Nuclear Company.

8) XN-NF-78-44(A), latest Revision, "Generic Control Rod Ejection Analysis," Exxon Nuclear Company.

9) XN-NF-621(A), latest Revision, "XNB Critical Heat Flux Correlation," Exxon Nuclear Company.

The acceptable Millstone 2 specific application of these analytical methodologies are described in ANF-88-126, "Millstone Unit 2 Cycle 10 Safety Analysis Report," dated October, 1988.

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, one copy to the Regional Administrator, Region I, and one copy to the NRC Resident Inspector within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

b. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

c. Safety Class 1 Inservice Inspection Program Review, Specification 4.4.10.1.

d. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

e. Fire Detection Instrumentation, Specifications 3.3.3.7.

f. Fire Suppression Systems, Specifications 3.7.9.1 and 3.7.9.2.

g. RCS Overpressure Mitigation, Specification 3.4.9.3

h. Radiological Effluent Reports required by Specifications 3.11.1.2, 3.11.2.2, 3.11.2.3 and 3.11.4.

i. Degradation of containment structure, Specification 4.6.1.6.4.

j. Steam Generator Tube Inspection, Specification 4.4.5.1.5.

k. Accident Monitoring Instrumentation, Specification 3.3.3.8.

l. Radiation Monitoring Instrumentation, Specification 3.3.3.1.

m. Reactor Coolant System Vents, Specification 3.4.11.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 148

TO FACILITY OPERATING LICENSE NO. DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

1.0 INTRODUCTION

By application for license amendment dated June 26, 1990 (Ref. 1), as amended by letter dated August 1, 1990 (Ref. 2), Northeast Nuclear Energy Company (the licensee) requested changes to the Technical Specifications (TS) for Millstone Nuclear Power Station, Unit 2. The proposed change would revise Technical Specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to a Core Operating Limits Report (COLR) for the values of those limits. The proposed changes also include the addition of the COLR to the Definitions section of the TS and to the reporting requirements of the Administrative Controls section of the TS. Guidance on the proposed changes was developed by NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket by Duke Power Company. This guidance was provided to all power reactor licensees and applicants by Generic Letter 88-16, dated October 4, 1988 (Ref. 3).

The August 1, 1990 submittal provided clarifications to the TS to provide consistency in the format, parameters and terminology of the TS. The changes did not alter the proposed action or affect the initial no significant hazards determination noticed in the Federal Register on July 25, 1990.

2.0 EVALUATION

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

(1) The definition section of the TS was modified to include a definition of the Core Operating Limits Report that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with NRC approved methodologies that maintain the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

(2) The following specifications were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

(a) Specification 3/4.1.1.1

With RCS T_{avg} greater than 200°F, the minimum shutdown margin for this specification is specified in the COLR.

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(b) Specification 3/4.1.1.2

With RCS T_{avg} less than or equal to 200°F, the minimum shutdown margin for this specification is specified in the COLR.

(c) Specification 3/4.1.1.4

The moderator temperature coefficient (MTC) limits for this specification are specified in the COLR and the upper limit still remains in the MTC technical specification.

(d) Specification 3/4.1.3.6

The regulating CEA insertion limits for this specification are specified in the COLR.

(e) Specification 3/4.2.1

The linear heat rate limits, including heat generated in the fuel, clad and moderator, for this specification are specified in the COLR.

(f) Specification 3/4.2.3

The total integrated radial peaking factor - F_r^T limits at rated thermal power for this specification are specified in the COLR.

(g) Specification 3/4.2.6

The limits for cold leg temperature, pressurizer pressure, reactor coolant flow rate and axial shape corresponding to the DNB margin for this specification are specified in the COLR.

The bases of affected specifications have been modified by the licensee to include appropriate reference to the COLR. Based on our review, we conclude that the changes to these bases are acceptable.

(3) Specification 6.9.1.7 is revised to add the Core Operating Limits Report to the reporting requirements of the Administrative Control section of the TS. This specification requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, these specifications require that the values of these limits be established using NRC approved methodologies and be consistent with all applicable limits of the safety analysis. The approved methodologies are the following:

(a) XN-75-27(A), latest Revisions and Supplements, "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," Exxon Nuclear Company.

(b) XN-NF-84-73(P), latest Revision and Supplements, "Exxon Nuclear Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Exxon Nuclear Company.

(c) XN-NF-82-21(A), latest Revision and Supplements, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company.

(d) XN-84-93(A), latest Revision and Supplements, "Steamline Break Methodology for PWR's," Exxon Nuclear Company.

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(f) XN-NF-82-49(A), latest Revision, "Exxon Nuclear Company Evaluation Model EXEM PWR Small Break Model," Exxon Nuclear Company.

(g) EXEM PWR Large Break LOCA Evaluation Model as defined by:

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XN-NF-82-07(A), latest Revision, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company.

XN-NF-81-58(A), latest Revision, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company.

XN-85-16(A), Volume 1 and Supplements, Volume 2 latest Revision and Supplements, "PWR 17x17 Fuel Cooling Test Program," Exxon Nuclear Company.

XN-NF-85-105(A), and Supplements, "Scaling of FCTF Based Reflood Heat Transfer Correlation for Other Bundle Designs," Exxon Nuclear Company.

(h) XN-NF-78-44(A), latest Revision, "Generic Control Rod Ejection Analysis," Exxon Nuclear Company.

Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC, prior to operation with the new parameter limits.

On the basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using NRC approved methodologies, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

As part of the implementation of Generic Letter 88-16, the staff has also reviewed a sample COLR that was provided by the licensee. On the basis of this review, the staff concludes that the format and content of the sample COLR are acceptable.

We have reviewed the request by the Northeast Nuclear Energy Company to modify the Technical Specifications of the Millstone Unit No. 2 that would remove the specific values of some cycle-dependent parameters from the specifications and place the values in a Core Operating Limits Report that would be referenced by the specifications. Based on this review, we conclude that these Technical Specification modifications are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The staff has previously published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). The amendment also involves changes to reporting or recordkeeping requirement. According, these changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

4.0 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 (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 REFERENCES

1. Letter (B13544) from E. J. Mroczka (NNECO) to NRC, dated June 26, 1990.

2. Letter (B13598) from E. J. Mroczka (NNECO) to NRC, dated August 1, 1990.

3. Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 4, 1988.

Dated: October 12, 1990

Principal Contributor:

T. Haung