June 28, 1989

Docket No. 50-423

Mr. Edward J. Mroczka
Senior Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
Post Office Box 270
Hartford, Connecticut 06141-0270

DISTRIBUTION

Docket File	EJordan, 3302 MNBB
NRC & Local PDRs	BGrimes, 9/A/2
Plant File	TMeek (4)
SVarga, 14/E/4	Wanda Jones
BBoger, 14/A/2	~~EButcher~~, 11/F/23 JCalvo
JStolz	ACRS (10)
SNorris	GPA/PA
DJaffe	ARM/LFMB
OGC (info. only)	WHodges
DHagan, 3302 MNBB	

Dear Mr. Mroczka:

SUBJECT: MILLSTONE UNIT 3 - ISSUANCE OF AMENDMENT (TAC NO. 71942)

The Commission has issued the enclosed Amendment No. 37 to Facility Operating License No. NPF-49 for Millstone Nuclear Power Station, Unit No. 3, in response to your application dated January 24, 1989.

The amendment changes the Millstone Unit 3 Technical Specifications (TS), to allow Cycle 3 operation, as follows: (1) TS 3/4.2.2 "Heat Flux Hot Channel Factor - Four Loops Operating and Three Loops Operating" would be changed to eliminate the reference to fuel assembly grid locations, (2) TS 5.3.1., "Fuel Assemblies" would be changed to properly describe the Cycle 3 fuel assemblies, (3) TS 5.3.2, "Control Rod Assemblies" would be changed to allow use of silver - indium - cadmium control rods and (4) TS 6.9.1.6, "Radial Peaking Factor Limit Report" would be changed to allow submittal of the report prior to each cycle's initial criticality.

A copy of the related Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

/s/

David H. Jaffe, Project Manager
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 37 to NPF-49
2. Safety Evaluation

cc w/enclosures:
See next page

[MILLSTONE 3 AMEND 71942]

LA:PDI-4	PM:PDI-4	BC:NRR/SRXB	D:PDI-4	OGC
SNorris	DJaffe:bd	WHodges	JStolz	MYoung
05/15/89	05/15/89	05/17/89	05/17/89	06/14/89

8907050106 890628
PDR ADOCK 05000423
P PDC

OFo1
1/1

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

June 28, 1989

Docket No. 50-423

Mr. Edward J. Mroczka
Senior Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
Post Office Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Mroczka:

SUBJECT: MILLSTONE UNIT 3 - ISSUANCE OF AMENDMENT (TAC NO. 71942)

The Commission has issued the enclosed Amendment No. 37 to Facility Operating License No. NPF-49 for Millstone Nuclear Power Station, Unit No. 3, in response to your application dated January 24, 1989.

The amendment changes the Millstone Unit 3 Technical Specifications (TS), to allow Cycle 3 operation, as follows: (1) TS 3/4.2.2 "Heat Flux Hot Channel Factor - Four Loops Operating and Three Loops Operating" would be changed to eliminate the reference to fuel assembly grid locations, (2) TS 5.3.1., "Fuel Assemblies" would be changed to properly describe the Cycle 3 fuel assemblies, (3) TS 5.3.2, "Control Rod Assemblies" would be changed to allow use of silver - indium - cadmium control rods and (4) TS 6.9.1.6, "Radial Peaking Factor Limit Report" would be changed to allow submittal of the report prior to each cycle's initial criticality.

A copy of the related Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

David H. Jaffe, Project Manager
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 37 to NPF-49
2. Safety Evaluation

cc w/enclosures:
See next page

cc:

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

W. D. Romberg, Vice President
Nuclear Operations
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

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

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

C. H. Clement, Unit Superintendent
Millstone Unit No. 3
Northeast Nuclear Energy Company
Post Office Box 128
Waterford, Connecticut 06385

Ms. Jane Spector
Federal Energy Regulatory Commission
825 N. Capitol Street, N.E.
Room 8608C
Washington, D.C. 20426

Burlington Electric Department
c/o Robert E. Fletcher, Esq.
271 South Union Street
Burlington, Vermont 05402

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

D. O. Nordquist
Manager of Quality Assurance
Northeast Nuclear Energy Company
Post Office Box 270
Hartford, Connecticut 06141-0270

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

First Selectmen
Town of Waterford
Hall of Records
200 Boston Post Road
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

M. R. Scully, Executive Director
Connecticut Municipal Electric
Energy Cooperative
30 Stott Avenue
Norwich, Connecticut 06360

Michael L. Jones, Manager
Project Management Department
Massachusetts Municipal Wholesale
Electric Company
Post Office Box 426
Ludlow, Massachusetts 01056

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

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

DOCKET NO. 50-423

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 37
License No. NPF-49

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 January 24, 1989, 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.

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. NPF-49 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 37 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its 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: June 28, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 37

FACILTIY OPERATING LICENSE NO. NPF-49

DOCKET NO. 50-423

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. The corresponding overleaf pages are provided to maintain document completeness.

Remove	Insert
3/4 2-9	3/4 2-9
3/4 2-13	3/4 2-13
5-5	5-5
6-21	6-21

2) When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.6;

f. The F_{xy} limits of Specification 4.2.2.1.2e., above, are not applicable in the following core planes regions as measured in percent of core height from the bottom of the fuel:

1) Lower core region from 0 to 15%, inclusive,

2) Upper core region from 85 to 100%, inclusive,

3) Within ±2% of core height (±2.88 inches) of grid plane regions. The total core height eliminated from the grid plane regions located between 15% and 85% of core height shall not exceed 20% of total core height (144 inches).

4) Core plane regions within ±2% of core height (± 2.88 inches) about the bank demand position of the Bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L} the effects of F_{xy} on $F_{Q}(Z)$ shall be evaluated to determine if $F_{Q}(Z)$ is within its limits.

4.2.2.1.3 When $F_{Q}(Z)$ is measured for other than F_{xy} determinations, an overall measured $F_{Q}(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

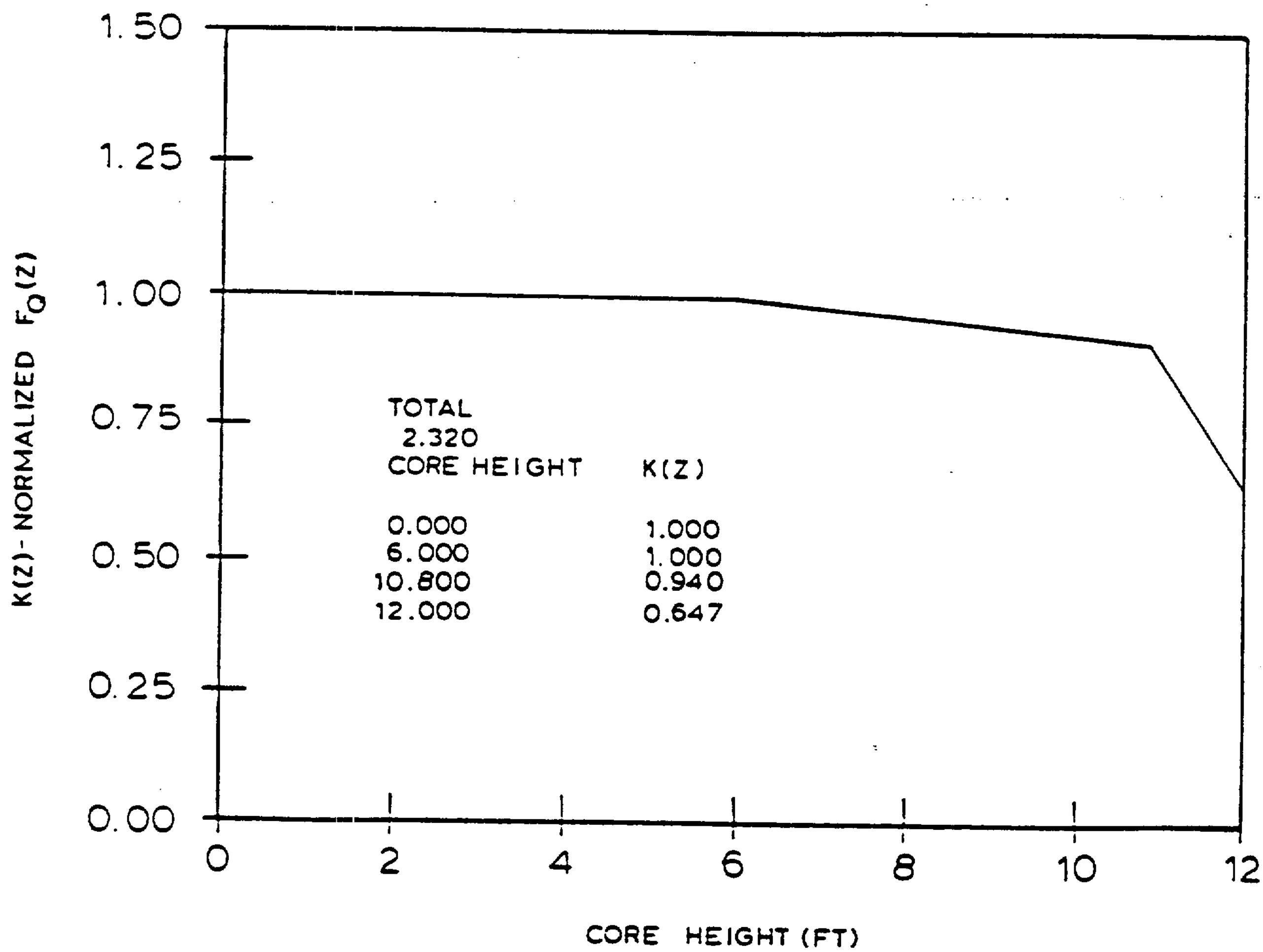

FIGURE 3.2-2a

K(Z) - NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT
FOR FOUR LOOP OPERATION

2) When the F_{xy}^{C} is less than or equal to the $F_{xy}^{0.65\ RTP}$ limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to $F_{xy}^{0.65\ RTP}$ and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for 65% of RATED THERMAL POWER ($F_{xy}^{0.65\ RTP}$) and the F_{xy} multiplier (M_{Fxy}) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.6;

f. The F_{xy} limits of Specification 4.2.2.2.2e., above, are not applicable in the following core planes regions as measured in percent of core height from the bottom of the fuel:

1) Lower core region from 0 to 15%, inclusive,

2) Upper core region from 85 to 100%, inclusive,

3) Within ±2% of core height (±2.88 inches) of grid plane regions. The total core height eliminated from the grid plane regions located between 15% and 85% of core height shall not exceed 20% of total core height (144 inches).

4) Core plane regions within ±2% of core height (±2.88 inches) about the bank demand position of the Bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L} the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

4.2.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determinations, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

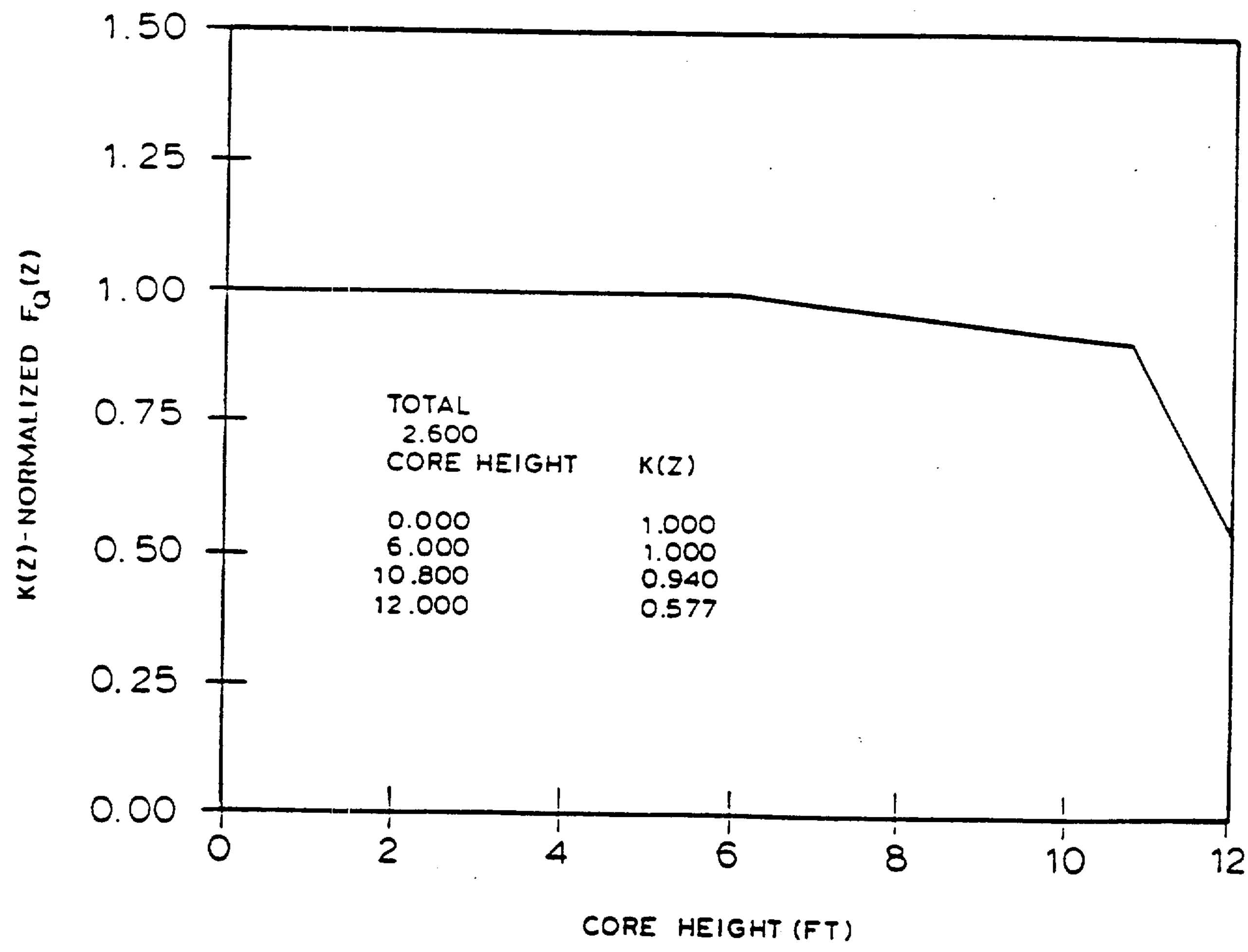

FIGURE 3.2-2b

K(Z) - NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT
FOR THREE LOOP OPERATION

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rod locations. Fuel rod locations may at any time during plant life have any combination of, 1) fuel rods clad with zircaloy-4, 2) filler rods fabricated from zircaloy-4 or stainless steel, or 3) vacancies, as determined by cycle-specific reload analysis. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum nominal enrichment of 3.4 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum nominal enrichment of 5.0 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 61 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 95.3% hafnium and 4.5% natural zirconium or 80% silver, 15% indium, and 5% cadmium. All control rods shall be clad with stainless steel.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,240 cubic feet at a nominal T_{avg} of 587°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-3.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 2.6% $\Delta k/k$ for uncertainties as described in Section 4.3 of the FSAR, and

b. A nominal 10.35-inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 45 feet.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 756 PWR fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.4 Routine Semiannual Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year.

A supplemental report containing dose assessments for the previous year shall be submitted annually within 90 days after January 1.

The report shall include that information delineated in the REMODCM.

Any changes to the REMODCM shall be submitted in the Semiannual Radioactive Effluent Release Report.

MONTHLY OPERATING REPORTS

6.9.1.5 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.

RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.6 The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided 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, for all core planes containing Bank "D" control rods and all unrodded core planes and the plot of predicted ($F_q^T \cdot P_{Rel}$) vs Axial Core Height with the limit envelope prior to each cycle initial criticality unless otherwise approved by the Commission by letter. In addition, in the event that the limit should change requiring a new substantial or an amended submittal to the Radial Peaking Factor Limit Report, it will be submitted prior to the date the limit would become effective unless otherwise approved by the Commission by letter. Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.

* A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

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.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS;

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications;

e. Records of changes made to the procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

h. Records of annual physical inventory of all sealed source material of record.

6.10.3 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories;

c. Records of radiation exposure for all individuals entering radiation control areas;

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO AMENDMENT NO. 37

TO FACILITY OPERATING LICENSE NO. NPF-49

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

By application for license amendment[1], Northeast Nuclear Energy Company, et al. (the licensee), requested changes to Millstone Unit 3 Technical Specifications (TS). The proposed amendment would change the Millstone Unit 3 TS to allow Cycle 3 operation as follows: (1) TS 3/4.2.2 "Heat Flux Hot Channel Factor - Four Loops Operating and Three Loops Operating" would be changed to eliminate the reference to fuel assembly grid locations, (2) TS 5.3.1., "Fuel Assemblies" would be changed to properly describe the Cycle 3 fuel assemblies, (3) TS 5.3.2, "Control Rod Assemblies" would be changed to allow use of silver - indium - cadmium control rods and (4) TS 6.9.1.6, "Radial Peaking Factor Limit Report" would be changed to allow submittal of the report prior to each cycle's initial criticality.

In response to a request from the NRC staff, the licensee submitted additional information[2,3] concerning Cycle 3 design and operation.

2.0 DESCRIPTION OF MILLSTONE 3, CYCLE 3

The Millstone Unit 3 reactor core is comprised of 193 fuel assemblies. The Cycle 3 core loading configuration features a low leakage pattern. During Cycle 2/3 refueling, 32 fresh Region 5A assemblies, 44 fresh Region 5B assemblies and 9 Region 2 assemblies from the spent fuel pool will replace 45 Region 2 fuel assemblies and 40 Region 3 fuel assemblies. A summary of the Cycle 3 fuel inventory is show below:

Region	2	3	4A	4B	5A	5B
Enrichment (w/o U-235)*	2.899	3.395	3.497	3.808	4.10**	4.50**
Geometric Density* (% theoretical)	94.965	94.980	95.13	95.17	95	95
Number of Assemblies	9	24	56	28	32	44
Approximate Burnup at Beginning o Cycle 3 (MWD/MTU)***	21,350	24,160	19,470	15,450	0	0

*All values are as-built except Region 5A and 5B
**Enrichment of enriched axial region of assemblies. Each Region 5 assembly also has six inches of 0.74 w/o axial blanket fuel at top and bottom.
***Based on actual EOC1 burnup of 18,700 MWD/MTU and nominal EOC2 burnup of 15,800 MWD/MTU.

2.1 Mechanical Design

The mechanical design of the Regions 5A and 5B fuel assemblies is the same as the Region 4 fuel assemblies except that the Region 5 assemblies will incorporate several upgraded fuel design features. These features include: (1) Extended Burnup Capability, (2) Reconstitutable Top Nozzles (RTNs), (3) Debris Filter Bottom Nozzles (DFBNs), (4) Integral Fuel Burnable Absorbers (IFBA), (5) Axial Blankets, and (6) Snag-Resistant Grids. These design improvements are described below.

(1) Extended Burnup Capability

The Region 5 fuel assembly design was modified for extended burnups by reducing the thickness of both the top nozzle and bottom nozzle end plates, decreasing the height of the top nozzle and bottom nozzle, and increasing the fuel rod length with a corresponding increase in the length of the fuel rod plenum. The fuel assembly overall height was adjusted to be consistent with fuel assembly growth predictions based upon accumulated Westinghouse in-core experience. This experience includes the results of high burnup demonstration programs conducted jointly by Westinghouse and utilities. These design changes allow for an additional distance between the nozzle plates, which is allocated for two purposes: (1) increased fuel rod growth associated with extended burnup and (2) increased fuel rod length to add plenum space for the increased fission gas release that occurs with increased burnup. As part of this design change, the grid elevations were relocated slightly to standardize the 17x17 fuel assembly design. Analyses have indicated the acceptability of the mechanical integrity of all fuel assembly components for extended burnup levels with the above changes. The methods and criteria established for Westinghouse fuel at extended burnup[4] have been approved by the NRC.

(2) Reconstitutable Top Nozzle (RTN)

The RTN differs from the current design in two ways: a groove is provided in each thimble thru-hole in the nozzle plate to facilitate attachment and removal; and the nozzle plate thickness was reduced to provide additional space for fuel rod growth. In conjunction with the RTN, a long tapered fuel rod bottom end plug is used to facilitate removal and reinsertion of the fuel rods. Details of the RTN design features, the design basis, and the evaluation of the RTN are given in Section 2.3.2 of Reference 4 which has been approved by the NRC.

(3) Debris Filter Bottom Nozzle (DFBN)

This bottom nozzle is designed to inhibit debris from entering the active fuel region of the core and thereby improves fuel performance by minimizing debris related fuel failures. The DFBN utilizes the same material, geometry, and welding requirements as its existing bottom nozzle counterpart. The DFBN is a low profile bottom nozzle design made of stainless steel, with reduced plate thickness and leg height thus providing additional space for fuel rod growth as part of the extended burnup feature. The DFBN is hydraulically equivalent to the existing bottom nozzle and meets all mechanical design functional requirements.

(4) Integral Fuel Burnable Absorber (IFBA)

The IFBA coated fuel pellets are identical to the enriched uranium dioxide pellets except for the addition of a thin boride coating on the pellet cylindrical surface along the central portion of the fuel stack length. IFBAs provide power peaking and moderator temperature coefficient control. Details of the IFBA design are given in Section 2.5 of Reference 4.

(5) Axial Blankets

The axial blanket consists of natural uranium (approximately 0.74 w/o) dioxide pellets at each end of the fuel stack to reduce neutron leakage and to improve uranium utilization. The axial blanket pellet design is the same as the enriched and IFBA pellet designs except for an increase in length. The length difference in the axial blanket pellets will help prevent accidental mixing with the enriched and IFBA pellets. Axial blankets are further discussed in Section 2.4 and 3.3 of Reference 4.

(6) Snag-Resistant Grids

The snag-resistant grids contain outer grid straps that are modified to help prevent assembly hangup due to grid strap interference during fuel assembly removal. This was accomplished by changing the grid strap corner geometry and adding guide tabs on the outer grid strap. Intermediate vanes to the top and tabs to the bottom of grids to reduce the potential of an assembly overlapping (and possibly locking) onto an adjacent fuel assembly. The corner chamfer is formed in the outside strap punching operation to eliminate grinding and the resultant sharp edge. In addition, a weld is placed on the small overlap on the top and bottom of the corners to increase strength and round over the leading edge of the corner.

In Cycle 3, some of the Hafnium RCCAs may be replaced with Ag-In-Cd Enhanced Performance Rod Cluster Control Assemblies (EP-RCCAs). The absorber diameter of the EP-RCCA is reduced slightly at the lower extremity of the rodlets in order to accommodate absorber swelling and minimize cladding interaction. However, the EP-RCCA design for Millstone Unit 3 does not include the wear resistance feature that is typically standard with the EP-RCCA design. The licensee is still evaluating the need for the wear resistance feature. With regard to the remaining hafnium control rods, the licensee is planning to conduct examinations of the hafnium control rod assemblies using the Westinghouse recommendations[5] during the next refueling outage which is scheduled to begin in May 20, 1989. Based on the results of the examinations, NNECO may elect to replace certain control rod assemblies with the silver-indium-cadmium rods or will provide a justification for continued operation with the existing hafnium control rod assemblies.

The Regions 5A and 5B fuel has been designed utilizing the latest Westinghouse fuel performance model[6], the Westinghouse clad flattening model[7], and the Westinghouse extended burnup methodology[8]. The Westinghouse fuel is designed and operated so that clad flattening will not occur for its planned residence time in the reactor. The fuel rod internal pressure design basis[9] is satisfied for all regions.

Westinghouse's experience with Zircaloy clad fuel is described in WCAP-8183, "Operational Experience with Westinghouse Cores[10]. This report is updated annually.

The principal design features of the Regions 5A and 5B fuel assemblies have been generically addressed by Westinghouse and accepted by the NRC staff. Based upon the above, we conclude that the use of the Regions 5A and 5B fuel assemblies is acceptable for Cycle 3 operation.

2.2 Nuclear Design

The nuclear design of the Cycle 3 core used Westinghouse codes approved by the NRC and the standard calculational methods described in the Westinghouse Reload Safety Evaluation Methodology[11]. This methodology is not affected by changes to the maximum uranium enrichment used in the fuel. The changes in physics characteristics for Cycle 3 are typical of the normal variations seen from cycle to cycle.

The Cycle 3 core loading is designed to meet a F_Q x P ECCS limit of $\leq$ 2.32 x K(Z)* for four loop operation and $\leq$ 2.60 x K(Z)* for three loop operation. The flux difference (ΔI) bandwidth during normal operation conditions +3, -12% for four loop operation and +5,-5% for three loop operation.

The Cycle 3 kinetics characteristics values fall within the current limits with the exception of the least negative Doppler temperature coefficient. There is no significant impact of this change in the least negative Doppler temperature coefficient, on the accident analysis (see Section 3.0, herein).

The licensee has addressed the control rod worths and requirements at the most limiting condition during the cycle for a core of 61 Hafnium rod cluster control assemblies (RCCAs). The required shutdown margin is based on previously submitted accident analysis[12]. The available shutdown margin exceeds the minimum required.

For Cycle 3 some Hafnium RCCAs may be replaced by Ag-In-Cd EP-RCCAs. This change has been evaluated to allow the exchange of the Ag-In-Cd EP-RCCAs for any number of Hf RCCAs provided that any control or shutdown bank consists entirely of only one type of absorber material. This is possible since both RCCA designs have similar neutronic characteristics. The largest change in total rod worth during the cycle is less than 100 pcm**. Core peaking factors change by less than 1%. As a result, the core performance characteristics of the Ag-In-Cd EP-RCCAs remain essentially the same. The available shutdown margin will exceed the minimum required shutdown margin and all other Technical

*K(Z) - See Figures 2 and 3 of Reference 3.
**pcm = 10^{-5}

Specification limits related to nuclear design will be met for any combination of Hf and Ag-In-Cd RCCAs in the configuration(s) described above.

2.3 Thermal and Hydraulic Design

No significant variations in thermal margins will result from the Cycle 3 reload. Sufficient DNB margin exists for all events to meet the design criteria[12,13] for the Cycle 3 reload core.

The DNB core limits and safety analysis used for Cycle 3 are based on conditions given in Sections 1.0 and 3.0. Fuel temperatures were calculated using the revised thermal safety model[14] and include the effects of standardized pellets.

3.0 ACCIDENT ANALYSIS

The licensee has evaluated the impact of the Cycle 3 reactor core design on the Millstone Unit 3, FSAR, Chapter 15 events. The following conclusions relate to the accident evaluation:

- For both large and small break LOCAs the Cycle 3 core configuration assures that the analysis presented in the FSAR remains bounding. Thus, operation during Cycle 3 meets the requirements of 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

- For non-LOCA events, with the exception of three events, the Chapter 15 FSAR analyses were determined to be bounding. The effects of Ag-In-Cd EP-RCCAs and the least negative Doppler Temperature Coefficient (inadvertent actuation of ECCS) were evaluated with regard to the Chapter 15 FSAR events and were found to be acceptable.

The three events that were reanalyzed by the licensee were the RCCA ejection, the steam system piping failure, and the reactor coolant pump shaft seizure (locked rotor).

3.1 RCCA Ejection Accidents

The RCCA ejection accident initiated from hot zero power (HZP) conditions at end-of-life (EOL) was reanalyzed for Cycle 3. The HZP EOL RCCA ejection case was the only case reanalyzed because the existing assumptions and results for the other cases continue to bound Cycle 3 operation. It should be noted the the HZP RCCA ejection event is only analyzed for N-loop operation since this analysis bounds the N-1 loop cases.

The results of the reanalysis demonstrated that the conclusions of the FSAR for the RCCA ejection event remain valid.

3.2 Steam System Piping Failure

The main steam line rupture event for N-loop operation was reanalyzed for Cycle 3 using revised core kinetics parameters. Both the limiting case that assumes the availability of offsite power throughout the event and the less severe case that includes a loss of offsite power were reanalyzed. No reanalysis was required for the N-1 loop main steamline rupture, since the existing analysis for that case continues to bound Cycle 3.

A DNB analysis was performed for the limiting case and it was determined that the conclusions of the FSAR for the main steam line rupture event remain valid. The DNB design basis, thus continues to be met for this event.

3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

Reanalysis of the locked rotor event was performed for Cycle 3 to predict the number of fuel rods that undergo DNB for N loop and N-1 loop operation. The analysis for N-1 loop operation assumed an initial nominal power level of 65% Rated Thermal Power (RTP). Previous analysis for this event, along with all other non-LOCA events, had assumed an initial nominal N-1 loop power level of 75% RTP. The actual licensed N-1 loop nominal power level is 65% RTP so that the use of 75% RTP represented a conservatism in the previous analysis. For Cycle 3, continued use of 75% RTP would have resulted in the predicted number of fuel rods that undergo DNB for this event exceeding the current limit value.

The result of the locked rotor reanalysis verified that less than 8.0% of the fuel rods were predicted to undergo DNB for the N-1 loop locked rotor event with the 65% RTP assumption. The reanalysis of the N loop locked rotor event verified that less than 6% of the fuel rods were predicted to undergo DNB. The radiological dose release evaluation for Cycle 3 was performed by the licensee. Doses were determined by adjusting the FSAR analysis doses to account for a higher number of fuel rods exceeding DNBR and to account for a shorter period of time to isolate the effected steam generator (from 30 minutes to 20 minutes). We find this approach to be acceptable.

The Cycle 3 locked rotor reanalysis described above was limited to the issue of determining the number of rods in DNB. The limiting cases of the current locked rotor licensing basis analysis intended to predict other transient conditions such as maximum RCS pressure, maximum clad temperature, and the magnitude of the zirconium steam reaction remain valid for Cycle 3. It should be noted that for N-1 Loop, use of the current locked rotor analysis means continuing to use an initial N-1 loop nominal power of 75% RTP. The reduced N-1 loop nominal power level of 65% RTP was only used for the rods in DNB analysis.

The results of the locked rotor analysis are acceptable and do not exceed the consequences of previous analyses.

4.0 DESIGN BASIS ACCIDENT ANALYSIS RELATIVE TO EXTENDED FUEL BURNUP

We have evaluated the potential impact of the radiological assessment of the design basis accidents (DBA), which were previously analyzed in the licensing of Millstone Unit 3.

An NRC publication entitled, "Assessment of the Use of Extended Burnup Fuel in Light Water Reactors," NUREG/CR 5009, February 1988, examined the changes that could result in the DBA assumptions, described in the various appropriate SRP sections and/or Regulatory Guides, that could result from the use of extended burnup fuel (up to 60,000 MWD/MT). The only DBA consequence that could be affected by the use of extended burnup fuel, even in a minor way, would be the potential thyroid doses that could result from a fuel handling accident.

NUREG-CR/5009 estimates that I-131 fuel gap activity in the peak fuel rod with 60,000 MWD/MT burnup could be as high as 12%. This value is approximately 20% higher than the 10% I-132 fuel gap activities value normally used in evaluating fuel handling accidents (Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facilities for Boiling and Pressurized Water Reactors").

We, therefore, reevaluated the fuel handling accident for Millstone Unit 3 with an increase in iodine gap activity in the fuel damage in a fuel handling accident. Listed below are the fuel handling accident thyroid doses presented in the operating licensing Safety Evaluation Report (NUREG-1031)[15], and the increased thyroid doses (by 20%) resulting from extended burnup fuel.

Exclusion Area Thyroid Dose (rem)		Low Population Zone Thyroid Dose (rem)	
NUREG-1031	20% Increase	NUREG-1031	20% Increase
1.8	2.2	0.1	0.1

The increased doses are still within our acceptance criterion stated in Standard Review Plan Section 15.7.4, i.e., well within the 300-rem thyroid exposure guidelines values set forth in 10 CFR Part 100.

5.0 TECHNICAL SPECIFICATIONS

The proposed amendment would change the Millstone Unit 3 Technical Specifications (TS) to allow Cycle 3 operation as follows: (1) TS 3/4.2.2 "Heat Flux Hot Channel Factor - Four Loops Operating and Three Loops Operating" would be change to eliminate the reference to fuel assembly grid locations, (2) TS 5.3.1., "Fuel Assemblies" would be changed to properly described the Cycle 3 fuel assemblies, (3) TS 5.3.2, "Control Rod Assemblies" would be change to allow use of silver - indium - cadmium control rods and (4) TS 6.9.1.6, "Radial Peaking Factor Limit Report" would be changed to allow submittal of the report prior to each cycle's initial criticality.

5.1 Fuel Assembly Grids

With regard to fuel assembly grids, TS 3/4.2.2 allows the licensee to exempt the measured values of the planar radial peaking factor (F_{xy}) when measured at the specified elevations where the grids are located. This is due to the effects of the grids which make the flux measurement inaccurate. Since the new fuel assemblies will have slightly off-set grid locations, the existing TS is not applicable. The licensee has proposed that the specific grid elevations in TS 4.2.2.1.2f.3 and 4.2.2.2.2f.3 be eliminated and replaced with a more general requirement that up to 20% of the core height, between 5% and 85% core height, can be eliminated from F_{xy} measurement.

As indicated in the licensee's application,[1] "The proposed change to delete specific grid plane centerline will not affect the total percent of the core monitored for F_{xy}. The previous requirement to monitor the core between 15% and 85% of core height except within ±2% of grid centerlines is essentially equal to the new requirement. There are 5 grids in the region to be monitored which amounts to 20% being excluded due to the effect of the grids and another 30% excluded at the top and bottom of the core. The total excluded area with the proposed requirement is also 20% due to grids and 30% at the top and bottom of the core". We concur with this assessment and conclude that the proposed changes to TS 4.2.2.1.2f.3 and 4.2.2.2.2f.3 are acceptable.

5.2 Fuel and Control Rod Design

The licensee has proposed changes to Section 5 of the TS. Changes to TS 5.3.1 and 5.3.2 would describe the Cycle 3 fuel assemblies and the use of silver-indium-cadmium control rods, respectively. While the proposed changes to TS 5.3.1 and 5.3.2 would permit the subject modifications to be used, any impact on reactor operation due to revised safety analysis results would be reflected in changes to the Limiting Conditions for Operation and/or the Surveillance Requirements; no such changes have been proposed for Cycle 3 except as described in Section 5.1. It should be noted that for proposed TS 5.3.1, although the TS and the safety analysis assume a maximum of 5.0 w/o fuel enrichment, the actual maximum Cycle 3 enrichment is 4.5 w/o enrichment. The changes described in proposed TS 5.3.1 and 5.3.2 represent refinements, rather than substantial changes, whose efficacy has been previously demonstrated in other operating facilities. We conclude that the proposed changes to TS 5.3.1 and 5.3.2 are acceptable.

5.3 Radial Peaking Factor Limit Report

The licensee has proposed a change to TS 6.9.1.6 to allow submittal of the Radial Peaking Factor Limit Report prior to initial-cycle criticality rather than 60 days prior to the time that the applicable limits become effective. The proposed change would allow the completion of cycle-specific calculations during the refueling outage when changes in core configuration, due to discovery of fuel leakage, may result. We conclude that the proposed change to TS 6.9.1.6 is acceptable.

6.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32 and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on June 27, 1989 (54 FR 27082). Accordingly, based upon the environ-mental assessment, we have determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

7.0 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.

8.0 REFERENCES

(1) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Proposed Revision to Technical Specifications," dated January 24, 1989.

(2) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Proposed Revision to Technical Specifications," dated February 23, 1989.

(3) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Cycle 3 Proposed Revision to Technical Specifications," dated April 12, 1989.

(4) Davidson, S.L., et al. "Reference Core Report VANTAGE 5 Fuel Assembly," WCAP-10444-P-A, September 1985.

(5) Johnson, W. J., "Hafnium RCCA Examination Guidelines," NS-NRC-89-3417, March 7, 1989

(6) Weiner, R. A. et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A, August 1988.

(7) George, R. A., et al., "Revised Clad Flattening Model," WCAP-8381, July 1974.

(8) Davidson, S. L. and Kramer, W. R., "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-R-A, December 1985.

(9) Risher, D. H., et al., "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis," WCAP-8964-A, August 1978.

(10) Foley, J., and Skaritka, J., "Operational Experience with Westinghouse Cores," (through December 31, 1987), WCAP-8183, Revision 16, August 1988.

(11) Davidson, S. L., et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9273-A, July 1985.

(12) "Final Safety Analysis Report Millstone Generating Station, Unit 3," USNRC Docket No. 50-423, December 1988.

(13) Letter from A. C. Thadani (NRC) to W. J. Johnson (Westinghouse), January 31, 1989, Subject: Acceptance for Referencing of Licensing Topical Report, WCAP-9226-P/9227-NP, "Reactor Core Response to Excessive Secondary Steam Releases."

(14) Leech, W. J., et al., "Revised PAD Code Thermal Safety Model," WCAP-8720, Addenda 2, October 1982.

(15) "Safety Evaluation Report Related to the Operation of Millstone Nuclear Power Station, Unit No. 3," Docket No. 50-423, July 1984.

Dated: June 28, 1989

Principal Contributor:

D. Jaffe