

FEB 22 1982

DCS MS-016

DISTRIBUTION:

- Docket File
- NRC PDR
- L PDR
- NSIC
- TERA
- ORB#3 Rdg
- DEisenhut
- PMKreutzer-3
- EConner
- RAClark
- OELD
- I&E-4
- GDeegan-4
- DMB

- DBrinkman
- ACRS-10
- CMiles
- RDiggs
- RBallard
- Chairman, ASLAB
- Gray File +4

Docket No. 50-336

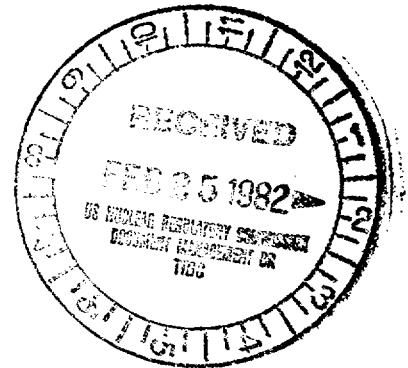
Mr. W. G. Council, Vice President
 Nuclear Engineering & Operations
 Northeast Nuclear Energy Company
 P. O. Box 270
 Hartford, Connecticut 06101

Dear Mr. Council:

The Commission has issued the enclosed Amendment No. 72 to Facility Operating Licence No. DPR-65 for Millstone Nuclear Power Station, Unit No. 2. This amendment consists of changes to the Technical Specifications (TS) in partial response to your application dated December 17, 1981.

The amendment changes the TS related to:

- o Mode 6 reactivity control,
- o Boration requirements,
- o Special test peak linear heat rate,
- o RTD response time testing,
- o Containment purge valves actuation,
- o ESFAS trip setpoints,
- o Core barrel monitoring,
- o Ventilation system HEPA filters, and
- o Editorial revisions.



Some portions of your proposed TS have been modified to meet our requirements. These modifications have been discussed with and agreed to by your staff. The remaining TS changes proposed in your application will be addressed in the Cycle 5 reload amendment to be issued later.

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

Sincerely,

Original signed by:

E. L. "Monte" Conner, Jr., Project Manager
 Operating Reactors Branch #3
 Division of Licensing

Enclosures:

1. Amendment No. 72 to DPR-65
2. Safety Evaluation
3. Notice of Issuance

*Conduct an
 amendment
 Fed Res. notice
 subject to
 typo in
 notice*

B203040064 B20222
 PDR ADOCK 05000336
 PDR

OFFICE	cc: See next page	ORB#3:DI PMKreutzer	ORB#3:DI EConner	ORB#3:DI RAClark	AD:OR:DL TNovak	OELD M. Rothschild
SURNAME						
DATE		2/5/82	2/5/82	2/5/82	2/8/82	2/12/82

Northeast Nuclear Energy Company

cc:

William H. Cuddy, Esquire
Day, Berry & Howard
Counselors at Law
One Constitution Plaza
Hartford, Connecticut 06103

Mr. Charles Brinkman
Manager - Washington Nuclear
Operations
C-E Power Systems
Combustion Engineering, Inc.
4853 Cordell Avn., Suite A-1
Bethesda, MD 20014

Mr. Lawrence Bettencourt, First Selectman
Town of Waterford
Hall of Records - 200 Boston Post Road
Waterford, Connecticut 06385

Northeast Nuclear Energy Company
ATTN: Superintendent
Millstone Plant
Post Office Box 128
Waterford, Connecticut 06385

Waterford Public Library
Rope Ferry Road, Route 156
Waterford, Connecticut 06385

U. S. Environmental Protection Agency
Region-I Office
ATTN: Regional Radiation
Representative
John F. Kennedy Federal Building
Boston, Massachusetts 02203

Northeast Utilities Service Company
ATTN: Mr. Richard T. Laudnat, Manager
Generation Facilities Licensing
P. O. Box 270
Hartford, Connecticut 06101

Mr. John Shedlosky
Resident Inspector/Millstone
c/o U.S.N.R.C.
P. O. Drawer KK
Niantic, CT 06357

Regional Administrator
Nuclear Regulatory Commission, Region I
Office of Inspection and Enforcement
631 Park Avenue
King of Prussia, Pennsylvania 19406

cc w/enclosure(s) and incoming
dtd: 12/17/81

Office of Policy & Management
ATTN: Under Secretary Energy
Division
80 Washington Street
Hartford, Connecticut 06115



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

NORTHEAST NUCLEAR ENERGY COMPANY
THE CONNECTICUT LIGHT AND POWER COMPANY
THE HARTFORD ELECTRIC LIGHT COMPANY
THE WESTERN MASSACHUSETTS ELECTRIC COMPANY

DOCKET NO. 50-336

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 72
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 December 17, 1981, 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.

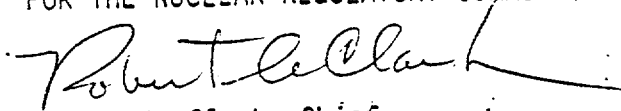
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-65 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 72, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 22, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 72

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

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

Pages

V	
X	3/4 7-16
1-7	3/4 7-17
3/4 1-1	3/4 7-18
3/4 1-3	3/4 9-1
3/4 1-16	3/4 9-16
3/4 2-3	3/4 9-17
3/4 3-1	3/4 9-18
3/4 3-4	3/4 10-1
3/4 3-12	B 3/4 1-3
3/4 3-14	B 3/4 4-12
3/4 3-18	B 3/4 5-1
3/4 3-19	B 3/4 6-2
3/4 3-20	B 3/4 9-1
3/4 3-24	
3/4 4-23	
3/4 4-24	
3/4 6-25	
3/4 6-26	
3/4 6-27	

INDEX

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>SECTION</u>	<u>PAGE</u>
3/4.4.4 PRESSURIZER.....	3/4 4-4
3/4.4.5 STEAM GENERATORS.....	3/4 4-5
3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE.....	3/4 4-8
Leakage Detection Systems.....	3/4 4-8
Reactor Coolant System Leakage.....	3/4 4-9
3/4.4.7 CHEMISTRY.....	3/4 4-10
3/4.4.8 SPECIFIC ACTIVITY.....	3/4 4-13
3/4.4.9 PRESSURE/TEMPERATURE LIMITS.....	3/4 4-17
Reactor Coolant System.....	3/4 4-17
Pressurizer.....	3/4 4-21
Overpressure Protection Systems.....	3/4 4-21a
3/4.4.10 STRUCTURAL INTEGRITY.....	3/4 4-22
 <u>3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)</u>	
3/4.5.1 SAFETY INJECTION TANKS.....	3/4 5-1
3/4.5.2 ECCS SUBSYSTEMS - $T_{avg} \geq 300^{\circ}\text{F}$	3/4 5-3
3/4.5.3 ECCS SUBSYSTEMS - $T_{avg} < 300^{\circ}\text{F}$	3/4 5-7
3/4.5.4 REFUELING WATER STORAGE TANK.....	3/4 5-8

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>SECTION</u>		<u>PAGE</u>
<u>3/4.6 CONTAINMENT SYSTEMS</u>		
3/4.6.1	PRIMARY CONTAINMENT.....	3/4 6-1
	Containment Integrity.....	3/4 6-1
	Containment Leakage.....	3/4 6-2
	Containment Air Locks.....	3/4 6-6
	Internal Pressure.....	3/4 6-8
	Air Temperature.....	3/4 6-9
	Containment Structural Integrity.....	3/4 6-10
3/4.6.2	DEPRESSURIZATION AND COOLING SYSTEMS.....	3/4 6-12
	Containment Spray System.....	3/4 6-12
	Containment Air Recirculation System.....	3/4 6-14
3/4.6.3	CONTAINMENT ISOLATION VALVES.....	3/4 6-15
3/4.6.4	COMBUSTIBLE GAS CONTROL.....	3/4 6-20
	Hydrogen Analyzers.....	3/4 6-20
	Electric Hydrogen Recombiners - <u>W</u>	3/4 6-21
	Hydrogen Purge System.....	3/4 6-23
	Post-Incident Recirculation Systems.....	3/4 6-24
3/4.6.5	SECONDARY CONTAINMENT.....	3/4 6-25
	Enclosure Building Filtration System.....	3/4 6-25
	Enclosure Building Integrity.....	3/4 6-28

INDEX

BASES

<u>SECTION</u>	<u>PAGE</u>
3/4.0 <u>APPLICABILITY</u>	B 3/4 0-1
<u>3/4.1 REACTIVITY CONTROL SYSTEMS</u>	
3/4.1.1 BORATION CONTROL.....	B 3/4 1-1
3/4.1.2 BORATION SYSTEMS.....	B 3/4 1-2
3/4.1.3 MOVABLE CONTROL ASSEMBLIES.....	B 3/4 1-3
<u>3/4.2 POWER DISTRIBUTION LIMITS</u>	
3/4.2.1 LINEAR HEAT RATE.....	B 3/4 2-1
3/4.2.2 TOTAL PLANAR RADIAL PEAKING FACTOR - F_{xy}^T	B 3/4 2-1
3/4.2.3 TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T	B 3/4 2-1
3/4.2.4 AZIMUTHAL POWER TILT.....	B 3/4 2-1
3/4.2.5 FUEL RESIDENCE TIME.....	B 3/4 2-2
3/4.2.6 DNB MARGIN.....	B 3/4 2-2
<u>3/4.3 INSTRUMENTATION</u>	
3/4.3.1 PROTECTIVE INSTRUMENTATION.....	B 3/4 3-1
3/4.3.2 ENGINEERED SAFETY FEATURE INSTRUMENTATION.....	B 3/4 3-1
3/4.3.3 MONITORING INSTRUMENTATION.....	B 3/4 3-2

INDEX

BASES

<u>SECTION</u>	<u>PAGE</u>
<u>3/4.4 REACTOR COOLANT SYSTEM</u>	
3/4.4.1 COOLANT LOOPS AND COOLANT CIRCULATION.....	B 3/4 4-1
3/4.4.2 SAFETY VALVES	B 3/4 4-1
3/4.4.3 RELIEF VALVES	B 3/4 4-2
3/4.4.4 PRESSURIZER	B 3/4 4-2
3/4.4.5 STEAM GENERATORS	B 3/4 4-2
3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE	B 3/4 4-3
3/4.4.7 CHEMISTRY	B 3/4 4-4
3/4.4.8 SPECIFIC ACTIVITY	B 3/4 4-4
3/4.4.9 PRESSURE/TEMPERATURE LIMITS	B 3/4 4-5
3/4.4.10 STRUCTURAL INTEGRITY	B 3/4 4-11
<u>3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)</u>	
3/4.5.1 SAFETY INJECTION TANKS	B 3/4 5-1
3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS	B 3/4 5-1
3/4.5.4 REFUELING WATER STORAGE TANK (RWST)	B 3/4 5-2

TABLE 1.1
OPERATIONAL MODES

<u>MODE</u>	<u>REACTIVITY CONDITION, K_{eff}</u>	<u>% RATED THERMAL POWER*</u>	<u>AVERAGE COOLANT TEMPERATURE</u>
1. POWER OPERATION	≥ 0.99	$> 5\%$	$\geq 300^{\circ}\text{F}$
2. STARTUP	≥ 0.99	$\leq 5\%$	$\geq 300^{\circ}\text{F}$
3. HOT STANDBY	< 0.99	0	$\geq 300^{\circ}\text{F}$
4. HOT SHUTDOWN	< 0.99	0	$300^{\circ}\text{F} > T_{avg}$ $> 200^{\circ}\text{F}$
5. COLD SHUTDOWN	< 0.98	0	$\leq 200^{\circ}\text{F}$
6. REFUELING**	≤ 0.95	0	$\leq 140^{\circ}\text{F}$

* Excluding decay heat.

** Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

TABLE 1.2
FREQUENCY NOTATION

<u>NOTATION</u>	<u>FREQUENCY</u>
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 6 months.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - $T_{avg} > 200^{\circ}\text{F}$

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be $\geq 3.20\% \Delta k/k$.

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

ACTION:

With the SHUTDOWN MARGIN $< 3.20\% \Delta k/k$, 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

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be $\geq 3.20\% \Delta k/k$:

- a. Immediately upon detection of an inoperable CEA. If the inoperable CEA is immovable or untrippable, the SHUTDOWN MARGIN, required by Specification 3.1.1.1, shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA.
- b. When in MODES 1 or 2, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.
- c. Prior to initial operation above 5% RATED THERMAL POWER after each refueling, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

* See Special Test Exception 3.10.1.

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - $T_{avg} \leq 200^{\circ}\text{F}$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be $\geq 2.0\%$ $\Delta k/k$.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN $< 2.0\%$ $\Delta k/k$, 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 $\geq 2.0\%$ $\Delta k/k$:

- a. Immediately upon detection of an inoperable CEA. If the inoperable CEA is immovable or untrippable, the SHUTDOWN MARGIN required by Specification 3.1.1.2 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.

REACTIVITY CONTROL SYSTEMS

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant through the core shall be ≥ 3000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.

APPLICABILITY: ALL MODES.

ACTION:

With the flow rate of reactor coolant through the core < 3000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The reactor coolant flow rate through the core shall be determined to be ≥ 3000 gpm prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

- a. Verifying at least one reactor coolant pump is in operation,
or
- b. Verifying that at least one low pressure safety injection pump is in operation and supplying ≥ 3000 gpm through the core.

REACTIVITY CONTROL SYSTEMS

BORIC ACID PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 The boric acid pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2a shall be OPERABLE if the flow path through the boric acid pump in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one boric acid pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2a inoperable, restore the boric acid pump to OPERABLE STATUS within 48 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 The boric acid pump(s) shall be demonstrated OPERABLE at least once per 7 days by:

- a. Starting (unless already operating) the pump from the control room,
- b. Verifying, that on recirculation flow, the pump develops a discharge pressure of ≥ 98 psig, and
- c. Verifying pump operation for at least 15 minutes.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

- a. One boric acid storage tank and one associated heat tracing circuit with the tank contents in accordance with Figure 3.1-1.
- b. The refueling water storage tank with:
 1. A minimum contained volume of 57,000 gallons,
 2. A minimum boron concentration of 1720 ppm when in Mode 5,
 3. A minimum boron concentration as defined in Specification 3.9.1 when in Mode 6.
 4. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

- a. At least once per 7 days by:
 1. Verifying the boron concentration of the water,
 2. Verifying the water level of the tank, and
 3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.
- b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the RWST ambient air temperature is < 35°F.

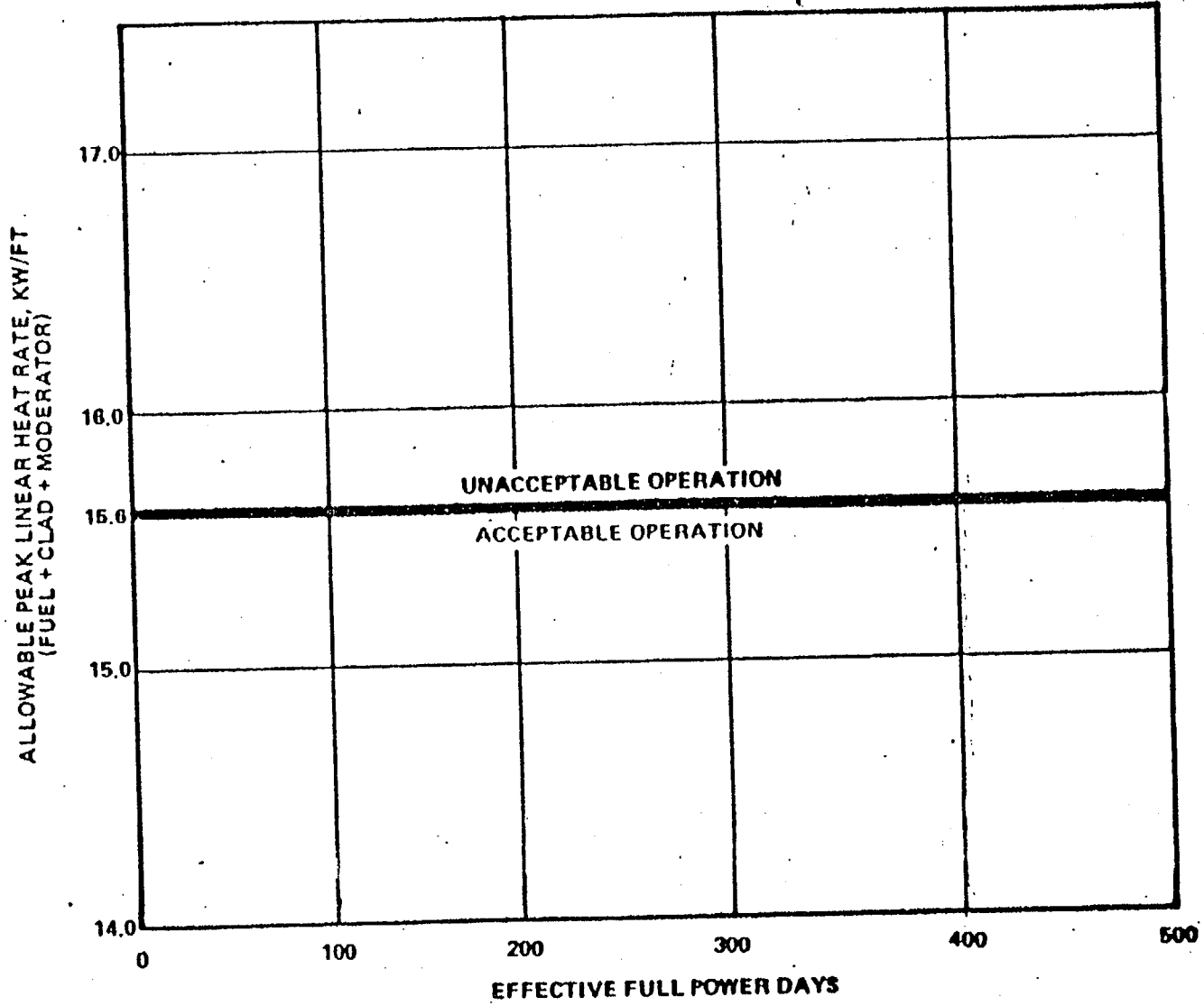


Figure 3.2-1 Allowable Peak Linear Heat Rate vs Burnup

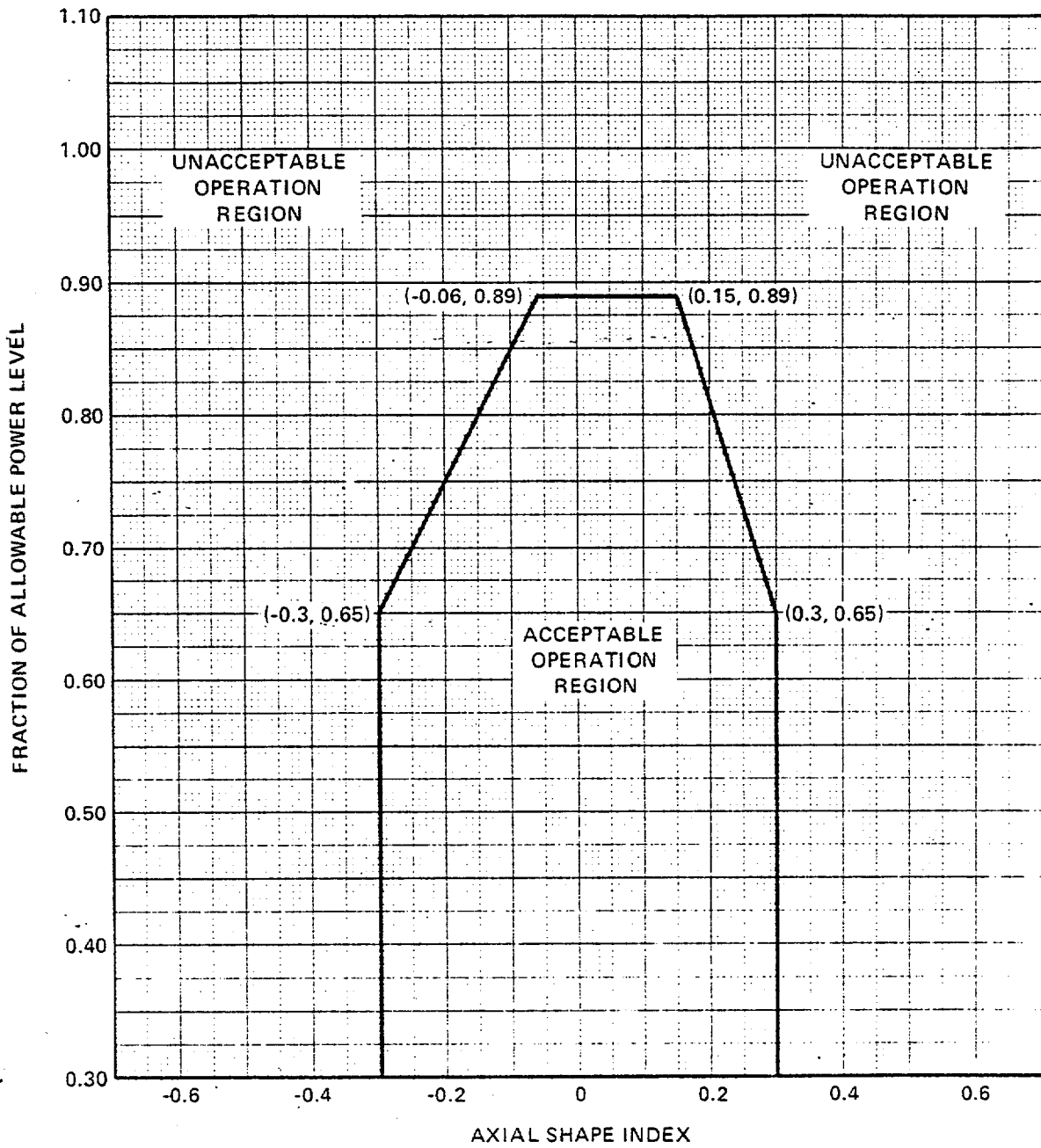


FIGURE 3.2-2 AXIAL SHAPE INDEX vs Fraction of Allowable Power Level per Specification 4.2.1.2c

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor protective instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

4.3.1.1.4 The response time of all REACTOR TRIP SYSTEM resistance temperature detectors (RTD) shall be verified to be less than or equal to the value specified in Table 3.3-2 within one month of operation for newly installed RTD's and once every 18 months thereafter.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

<u>FUNCTIONAL UNIT</u>	<u>TOTAL NO. OF CHANNELS</u>	<u>CHANNELS TO TRIP</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ACTION</u>
1. Manual Reactor Trip	2	1	2	1, 2 and *	1
2. Power Level - High	4	2 (f)	3	1, 2	2
3. Reactor Coolant Flow - Low	4	2(a)	3	1, 2 (e)	2
4. Pressurizer Pressure - High	4	2	3	1, 2	2
5. Containment Pressure - High	4	2	3	1, 2	2
6. Steam Generator Pressure - Low	4	2(b)	3	1, 2	2
7. Steam Generator Water Level - Low	4	2	3	1, 2	2
8. Local Power Density - High	4	2(c)	3	1	2
9. Thermal Margin/Low Pressure	4	2(a)	3	1, 2 (e)	2
10. Loss of Turbine--Hydraulic Fluid Pressure - Low	4	2(c)	3	1	3

MILLSTONE - UNIT 2

3/4 3-2

Amendment No. 15

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

<u>FUNCTIONAL UNIT</u>	<u>TOTAL NO. OF CHANNELS</u>	<u>CHANNELS TO TRIP</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ACTION</u>
11. Wide Range Logarithmic Neutron Flux Monitor - Shutdown	4	0	2	3, 4, 5	4
12. Underspeed - Reactor Coolant Pumps	4	2(a)	3	1, 2(e)	2

MILLSTONE - UNIT 2

3/4 3-3

Amendment No. 15, 28, 52

TABLE 3.3-1 (Continued)

TABLE NOTATION

* With the protective system trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

- (a) Trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is \geq 5% of RATED THERMAL POWER.
- (b) Trip may be manually bypassed below 600 psia; bypass shall be automatically removed at or above 600 psia.
- (c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is \geq 15% of RATED THERMAL POWER.
- (d) Deleted.
- (e) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.
- (f) ΔT Power input to trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is \geq 5% of RATED THERMAL POWER.

ACTION STATEMENTS

- ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 4 hours and/or open the protective system trip breakers.
- ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels and with the THERMAL POWER level:
 - a. $<$ 5% of RATED THERMAL POWER, immediately place the inoperable channel in the bypassed condition; restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER.
 - b. $>$ 5% of RATED THERMAL POWER, operation may continue with the inoperable channel in the bypassed condition, provided the following conditions are satisfied:

INSTRUMENTATION

SURVEILLANCE REQUIREMENTS (Continued)

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESF function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESF function as shown in the "Total No. of Channels" Column of Table 3.3-3.

4.3.2.1.4 The trip value shall be such that the containment purge effluent shall not result in calculated concentrations of radioactivity offsite in excess of 10 CFR Part 20, Appendix B, Table II.6. For the purposes of calculating this trip value, a $x/Q = 5.8 \times 10^{-6} \text{ sec/m}^3$ shall be used when the system is aligned to purge through the building vent and a $X/Q = 7.5 \times 10^{-8} \text{ sec/m}^3$ shall be used when the system is aligned to purge through the Unit 1 stack, the gaseous and apticulate (Half Lives greater than 8 days) radioactivity shall be asumed to be Xe-133 and Cs-137, respectively. However, the setpoints shall be no greater than $5 \times 10^5 \text{ cpm}$.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

<u>FUNCTIONAL UNIT</u>	<u>TOTAL NO. OF CHANNELS</u>	<u>CHANNELS TO TRIP</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ACTION</u>
1. SAFETY INJECTION (SIAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	1
b. Containment Pressure - High	4	2	3	1, 2, 3	2
c. Pressurizer Pressure - Low	4	2	3	1, 2(e), 3(a)	2
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	1
b. Containment Pressure -- High - High	4	2(b)	3	1, 2, 3	2
3. CONTAINMENT ISOLATION (CIAS)					
a. Manual CIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	1
b. Manual SIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	1
c. Containment Pressure - High	4	2	3	1, 2, 3	2
d. Pressurizer Pressure - Low	4	2	3	1, 2(e), 3(a)	2

MILLSTONE - UNIT 2

3/4 3-12

Amendment No. 72

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

<u>FUNCTIONAL UNIT</u>	<u>TOTAL NO. OF CHANNELS</u>	<u>CHANNELS TO TRIP</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ACTION</u>
4. MAIN STEAM LINE ISOLATION					
Steam Generator Pressure - Low	4	2	3	1, 2, 3(c)	2
5. ENCLOSURE BUILDING FILTRATION (EBFAS)					
a. Manual EBFAS (Trip Buttons)	2	1	2	1, 2, 3, 4	1
b. Manual SIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	1
c. Containment Pressure - High	4	2	3	1, 2, 3	2
d. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	2
6. CONTAINMENT SUMP RECIRCULATION (SRAS)					
a. Manual SRAS (Trip Buttons)	2	1	2	1, 2, 3, 4	1
b. Refueling Water Storage Tank - Low	4	2	3	1, 2, 3	2

MILLSTONE - UNIT 2

3/4 3-13

Amendment No. 15, 63

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

<u>FUNCTIONAL UNIT</u>	<u>TOTAL NO. OF CHANNELS</u>	<u>CHANNELS TO TRIP</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ACTION</u>
7. CONTAINMENT PURGE VALVE ISOLATION					
a. Containment Radiation - High				5, 6	
Gaseous Monitor	1(d)	1(d)	1		3
Particulate Monitor	1(d)	1(d)	1		3
8. LOSS OF POWER					
a. 4.16 kv Emergency Bus Undervoltage (Undervoltage relays) - level one	4/bus	2/Bus	3/bus	1, 2, 3	2
b. 4.16 kv Emergency Bus Undervoltage (Undervoltage relays) - level two	4/Bus	2/Bus	3/Bus	1, 2, 3	2

MILLSTONE - UNIT 2

3/4 3-14

Amendment No. 88, 72

TABLE 3.3-3 (Continued)

ACTION 3 - With one or more channels inoperable, operation may continue provided the containment purge valves are maintained closed.

MILLSTONE - UNIT 2

3/4 3-18

Amendment No. 72

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 4.75 psig	≤ 5.20 psig
c. Pressurizer Pressure - Low	≥ 1600 psia	≥ 1592.5 psia
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High-High	≤ 27 psig	≤ 27.45 psig
3. CONTAINMENT ISOLATION (CIAS)		
a. Manual CIAS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Containment Pressure - High	≤ 4.75 psig	≤ 5.20 psig
d. Pressurizer Pressure - Low	≥ 1600 psia	≥ 1592.5 psia
4. MAIN STEAM LINE ISOLATION		
Steam Generator Pressure - Low	≥ 500 psia	≥ 492.5 psia

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

<u>FUNCTIONAL UNIT</u>	<u>TRIP VALUE</u>	<u>ALLOWABLE VALUES</u>
5. ENCLOSURE BUILDING FILTRATION (EBFAS)		
a. Manual EBFAS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Containment Pressure - High	≤ 4.75 psig	≤ 5.20 psig
d. Pressurizer Pressure - Low	≥ 1600 psia	≥ 1592.5 psia
6. CONTAINMENT SUMP RECIRCULATION (SRAS)		
a. Manual SRAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Storage Tank - Low	48 ± 9 inches above tank bottom	48 ± 18 inches above tank bottom
7. CONTAINMENT PURGE VALVES ISOLATION		
a. Containment Radiation - High		
Gaseous Activity	$<$ the value determined in accordance with Specification 4.3.2.1.4.	$<$ the value determined in accordance with Specification 4.3.2.1.4.
Particulate Activity (Half Lives greater than 8 days)	$<$ the value determined in accordance with Specification 4.3.2.1.4.	$<$ the value determined in accordance with Specification 4.3.2.1.4.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

<u>FUNCTIONAL UNIT</u>	<u>TRIP VALUE</u>	<u>ALLOWABLE VALUES</u>
8. LOSS OF POWER		
a. 4.16 kv Emergency Bus Undervoltage (Undervoltage relays) - level one	≥ 2912 volts	≥ 2877 volts
b. 4.16 kv Emergency Bus Undervoltage (Undervoltage relays) - level two	≥ 3700 volts with an 8.0 + 2.0 second time delay	≥ 3663 volts with an 8.0 + 2.0 second time delay
9. AUXILIARY FEEDWATER		
a. Manual	Not Applicable	Not Applicable
b. Steam Generator Level - Low	$\geq 12\%$	$\geq 10\%$

MILLSTONE - UNIT 2

3/4 3-20

Amendment No. 18, 63, 72

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>MODES IN WHICH SURVEILLANCE REQUIRED</u>
1. SAFETY INJECTION (SIAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Containment Pressure - High	S	R	M	1, 2, 3
c. Pressurizer Pressure - Low	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
2. CONTAINMENT SPRAY (CSAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Containment Pressure -- High - High	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
3. CONTAINMENT ISOLATION (CIAS)				
a. Manual CIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
c. Containment Pressure - High	S	R	M	1, 2, 3
d. Pressurizer Pressure - Low	S	R	M	1, 2, 3
e. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
4. MAIN STEAM LINE ISOLATION				
a. Steam Generator Pressure - Low	S	R	M	1, 2, 3
b. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
5. ENCLOSURE BUILDING FILTRATION (EBFAS)				
a. Manual EBFAS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
c. Containment Pressure - High	S	R	M	1, 2, 3
d. Pressurizer Pressure - Low	S	R	M	1, 2, 3
e. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3

MILLSTONE - UNIT 2

3/4 3-24

Amendment No. 63, 72

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>MODES IN WHICH SURVEILLANCE REQUIRED</u>
6. CONTAINMENT SUMP RECIRCULATION (SRAS)				
a. Manual SRAS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Refueling Water Storage Tank - Low	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
7. CONTAINMENT PURGE VALVES ISOLATION				
a. Containment Radiation - High Gaseous Monitor	S	R	M	ALL MODES
Particulate Monitor	S	R	M	ALL MODES
8. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Undervoltage relays) - level one	S	R	M	1, 2, 3
b. 4.16 kv Emergency Bus Undervoltage (Undervoltage relays) - level two	S	R	M	1, 2, 3
9. AUXILIARY FEEDWATER				
a. Manual	N.A.	N.A.	R	N.A.
b. Steam Generator Level - Low	S	R	M	1, 2, 3

This page intentionally left blank

This page intentionally left blank

CONTAINMENT SYSTEMS

3/4.6.5 SECONDARY CONTAINMENT

ENCLOSURE BUILDING FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.1 Two separate and independent enclosure building filtration systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one enclosure building filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in COLD SHUT-DOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 Each enclosure building filtration system shall be demonstrated OPERABLE:

- a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 10 hours with the heaters on.
- b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 9000 cfm \pm 10%.
 2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.
 3. Verifying a system flow rate of 9000 cfm \pm 10% during system operation when tested in accordance with ANSI N510-1975.
- c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.
- d. At least once per 18 months by:
1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is \leq 6 inches Water Gauge while operating the system at a flow rate of 9000 cfm \pm 10%.
 2. Verifying that the system starts on an Enclosure Building Filtration Actuation Signal (EBFAS).
 3. Verifying that each system produces a negative pressure of greater than or equal to 0.25 inches W.G. in the Enclosure Building Filtration Region within (1) minute after an EBFAS.
- e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 9000 cfm \pm 10%.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 9000 cfm \pm 10%.

CONTAINMENT SYSTEMS

ENCLOSURE BUILDING INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.2 ENCLOSURE BUILDING INTEGRITY shall be maintained.

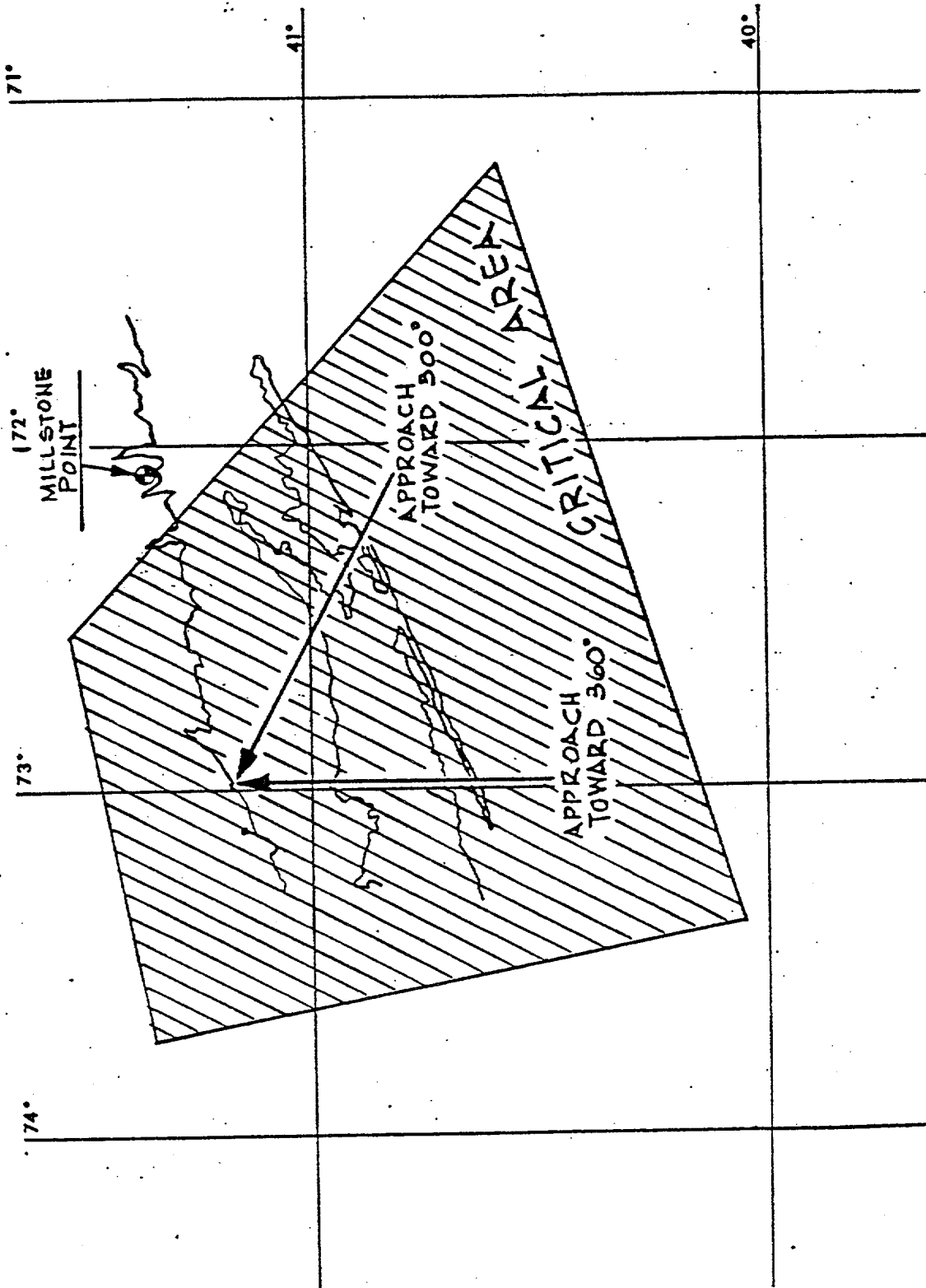
APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without ENCLOSURE BUILDING INTEGRITY, restore ENCLOSURE BUILDING INTEGRITY within 24 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.2 ENCLOSURE BUILDING INTEGRITY shall be demonstrated at least once per 31 days by verifying that each door in each access opening is closed except when the access opening is being used for normal transit entry and exit.



Critical Area
Figure 3.7-1

PLANT SYSTEMS

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 Two independent control room emergency ventilation systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one control room emergency ventilation system inoperable, restore the system to OPERABLE status within 7 days or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.7.6.1 Each control room emergency ventilation system shall be demonstrated OPERABLE:

- a. At least once per 12 hours by verifying that the control room air temperature is $\leq 120^{\circ}\text{F}$.
- b. At least once per 31 days on a STAGGERED TEST BASIS by initiating from the control room, flow through the HEPA filters and charcoal absorber train and verifying that the system operates for at least 15 minutes.
- c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 2000 cfm \pm 10%.
2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.
3. Verifying a system flow rate of 2000 cfm \pm 10% during system operation when tested in accordance with ANSI N510-1975.
- d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.
- e. At least once per 18 months by:
 1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gause while operating the system at a flow rate of 2000 cfm \pm 10%.
 2. Verifying that on a recirculation signal, the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.
- f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm \pm 10%.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm \pm 10%.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATIONS

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of following reactivity conditions is met:

- a. Either a K_{eff} of 0.95 or less, or
- b. A boron concentration of greater than or equal to 1720 ppm.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, within 15 minutes suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 40 gpm of boric acid solution at or greater than the required refueling water storage tank concentration (ppm) until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 1720 ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

- a. Removing or unbolting the reactor vessel head, and
- b. Withdrawal of any full length CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of all filled portions of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

*The reactor shall be maintained in MODE 6 whenever the reactor vessel head is unbolted or removed and fuel is in the reactor vessel.

REFUELING OPERATIONS

INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be operating, each with continuous visual indication in the control room and one with audible indication in the containment.

APPLICABILITY: MODE 6.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

- a. A CHANNEL FUNCTIONAL TEST at least once per 7 days.
- b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the start of CORE ALTERATIONS, and
- c. A CHANNEL CHECK at least once per 12 hours during CORE ALTERATIONS.

TABLE 3.9-1

ACCESS DOORS TO SPENT FUEL POOL AREA

<u>Door No.</u>	<u>Elevation</u>	<u>Location</u>	<u>Type</u>	<u>Area Serviced</u>
291	14'6"	M.7 - 18.5	Double Door	SFP Skimmer System
292 or 207	14'6"	R/S - 18.9	Double Door 8' Rollup Door ^{or}	Solidification System
293	14'6"	Q/R - 18.0	Double Door	Maintenance Shop
208	14'6"	S - 18.9	16' Rollup Door	Railway Access
294	14'6"	Q - 20.7	Single Door	D/G Room
295	38'6"	F.8 - 18	8' Rollup Door	Aux. & R. W. HVAC
296	38'6"	F.8 - 18.5	Single Door	Aux. & R. W. HVAC
297	38'6"	F.8 - 18.5	Single Door	North Stairwell
---	38'6"	H.4 - 18.9	Double Sliding Door	Elevator
298	38'6"	M.4 - 18.9	Single Door	Penetration Room
299	38'6"	M.7 - 18.9	Double Door	Main Exh. Fan Room
247	38'6"	M.7 - 17.2	Single Door	South Stairwell
254	55'6"	S - 17.2	Single Door	Roof Above Storage Floor
253	55'6"	S - 18.9	Single Door	Roof Above F. O. Tanks

REFUELING OPERATIONS

STORAGE POOL AREA VENTILATION SYSTEM - FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.15 At least one Enclosure Building Filtration System shall be OPERABLE and capable of automatically initiating operation in the auxiliary exhaust mode and exhausting through HEPA filters and charcoal adsorbers on a storage pool area high radiation signal.

APPLICABILITY: WHENEVER IRRADIATED FUEL IS IN THE STORAGE POOL.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one spent fuel storage pool ventilation system is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.9.15 The above required Enclosure Building Filtration System shall be demonstrated OPERABLE:

- a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.
- b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS. (Continued)

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 9000 cfm \pm 10%.
 2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.5.a of Regulatory Guide 1.52, Revision 2, March 1978.
 3. Verifying a system flow rate of 9000 cfm \pm 10% during system operation when tested in accordance with ANSI N510-1975.
- c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.
- d. At least once per 18 months by:
1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is \leq 6 inches Water Gauge while operating the system at a flow rate of 9000 cfm \pm 10%.
 2. Verifying that on a Spent Fuel Storage Pool Area high radiation signal, the system automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks.
- e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 9000 cfm \pm 10%.

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

- f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 9000 cfm \pm 10%.

3/4.10 SPECIAL TEST EXCEPTIONS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODES 2 and 3.

ACTION:

- a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, 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 SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.
- b. With all full length CEAs inserted and the reactor subcritical by less than the above reactivity equivalent, immediately 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 SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

SPECIAL TEST EXCEPTIONS

GROUP HEIGHT AND INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

- a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and
- b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1, being exceeded while the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 and 3.2.4 are suspended, immediately:

- a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1 or
- b. Be in HOT STANDBY within 2 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 or 3.2.4 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.3 and 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 or 3.2.4 are suspended.

REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.2 BORATION SYSTEMS (Continued)

The boron capability required below 200°F is based upon providing a 2% $\Delta k/k$ SHUTDOWN MARGIN at 140°F during refueling with all full and part length control rods withdrawn. This condition requires either 5,050 gallons of 6.25% boric acid solution from the boric acid tanks or 57,000 gallons of 1720 ppm borated water from the refueling water storage tank.

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.

3/4.1.3 MOVABLE 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 of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a immovable or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (< 20 steps) of the CEAs, there is 1) a small degradation in the peaking factors relative to those assumed in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 2) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 3) a small effect on the available SHUTDOWN MARGIN, and 4) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the small misalignment of a CEA permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements prior to initiating a reduction in THERMAL POWER. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

Overpower margin is provided to protect the core in the event of a large misalignment (> 20 steps) of a CEA. However, this misalignment would cause distortion of the core power distribution. The reactor

REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.3 MOVABLE CONTROL ASSEMBLIES (Continued)

protective system would not detect the degradation in radial peaking factors and since variations in other system parameters (e.g., pressure and coolant temperature) may not be sufficient to cause trips, it is possible that the reactor could be operating with process variables less conservative than those assumed in generating LCO and LSSS setpoints. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt and significant reduction in THERMAL POWER prior to attempting realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of the CEA position indicators (Specification 3.1.3.3) is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits and ensures proper operation of the rod block circuit. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The maximum CEA drop time permitted by Specification 3.1.3.4 is the assumed CEA drop time used in the accident analyses. Measurement with $T_{avg} \geq 515^{\circ}\text{F}$ and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

REACTOR COOLANT SYSTEM

BASES

for piping, pumps and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-3 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs or an RCS vent opening of greater than 1.3 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are $\leq 275^{\circ}\text{F}$. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator $\leq 43^{\circ}\text{F}$ (31°F when measured by a surface contact instrument) above the coolant temperature in the reactor vessel or (2) the start of a HPSI pump and its injection into a water solid RCS.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

This page intentionally left blank.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the RCS safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration and pressure ensure that the assumptions used for safety injection tank injection in the accident analysis are met.

The limit of one hour for operation with an inoperable safety injection tank minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward.

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provides this protection by dissolving in the sump water and causing its final pH to be raised to ≥ 7.0 . This determination assumes the RCS, the SI tanks, and the RWST is at a maximum boron concentration of 2400 ppm.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the accident analyses are met and that subsystem OPERABILITY is maintained. The requirement to dissolve a representative sample of TSP in a sample of RWST water provides assurance that the stored TSP will dissolve in borated water at the postulated post LOCA temperatures. The ECCS leak rate surveillance requirements assure that the leakage rates assumed for the system outside containment during the recirculation phase will not be exceeded.

EMERGENCY CORE COOLING SYSTEMS

BASES

The purpose of the ECCS throttle valve surveillance requirements is to provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 REFUELING WATER STORAGE TANK (RWST)

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 54 psig, P_a . As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq 0.75 L_a$ during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and leak rate given in Specifications 3.6.1.1 and 3.6.1.2. The limitations on the air locks allow entry and exit into and out of the containment during operation and ensure through the surveillance testing that air lock leakage will not become excessive through continuous usage.

CONTAINMENT SYSTEMS

BASES

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that the containment peak pressure does not exceed the design pressure of 54 psig during LOCA conditions.

The maximum peak pressure obtained from a LOCA event is 53.8 psig. The limit of 2.1 psig for initial positive containment pressure will limit the total pressure to less than the design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment air temperature ensures that the containment peak air temperature does not exceed the design temperature of 288°F during LOCA conditions. The containment temperature limit is consistent with the accident analyses.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 53.8 psig in the event of a LOCA. The measurement of containment tendon lift off force, the visual and metallurgical examination of tendons, anchorages and liner and the Type A leakage tests are sufficient to demonstrate this capability.

The surveillance requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Regulatory Guide 1.35 "Inservice Surveillance of UngROUTED Tendons in Prestressed Concrete Containment Structures".

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during fuel or CEA movement within the reactor pressure vessel.

REFUELING OPERATIONS

BASES

3/4.9.6 CRANE OPERABILITY - CONTAINMENT BUILDING

The OPERABILITY requirements of the cranes used for movement of fuel assemblies ensures that: 1) each crane has sufficient load capacity to lift a fuel element, and 2) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel assembly and CEA over irradiated fuel assemblies ensures that no more than the contents of one fuel assembly will be ruptured in the event of a fuel handling accident. This assumption is consistent with the activity release assumed in the accident analyses.

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

The requirement that at least one shutdown cooling loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

The requirement to have two shutdown cooling loops OPERABLE when the refuel pool is unavailable as a heat sink ensures that a single failure of the operating shutdown cooling loop will not result in a complete loss of decay heat removal capability. With the reactor vessel water level at or above the vessel flange, the reactor vessel pit seal installed, and a combined available volume of water in the refueling pool and refueling water storage tank in excess of 370,000 gallons, a large heat sink is readily available for core cooling. Adequate time is thus available to initiate emergency procedures to provide core cooling in the event of a failure of the operating shutdown cooling loop.

3/4.9.9 and 3/4.9.10 CONTAINMENT RADIATION MONITORING AND CONTAINMENT PURGE VALVE ISOLATION SYSTEM

The OPERABILITY of these systems ensures that the containment purge valves will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of these systems is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.11 and 3/4.9.12 WATER LEVEL-REACTOR VESSEL AND STORAGE POOL WATER LEVEL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 72 TO FACILITY OPERATING LICENSE NO. DPR-65
NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.
MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2
DOCKET NO. 50-336

Introduction

By application dated December 17, 1981, Northeast Nuclear Energy Company (NNECO or the licensee) proposed Technical Specification (TS) changes necessary for the Cycle 5 operation of Millstone Unit No. 2. In addition to those TS necessary for the next cycle of operation, numerous other TS requests were submitted. To reduce the size of the reload package, we have determined that the following proposed changes may be evaluated independent of the Cycle 5 reload.

- Mode 6 reactivity control,
- Boration requirements,
- Special test peak linear heat rate,
- RTD response time testing,
- Containment purge valves actuation,
- ESFAS allowable valves,
- Core barrel monitoring,
- Ventilation system HEPA filters, and
- Editorial revisions.

Discussion and Evaluation

Each of the above proposed TS changes will be discussed and evaluated in the subsequent sections of this safety evaluation.

Mode 6 Reactivity Control

NNECO proposed changes to TS Table 1.1 and TS 3.1.2.7, 3.9.1 and B 3/4.9.1. to redefine the Mode 6 reactivity control. The current TS Table 1.1 defines Mode 6 (refueling) reactivity conditions (K_{eff}) as < 0.90 . TS 3.9.1 requires all full length control element assemblies (CEAs) to be fully inserted. The proposed change would increase the refueling K_{eff} to < 0.95 , remove the requirement for all CEA insertion and specify an acceptable boron concentration of > 1720 ppm. This is necessary to be consistent with the initial conditions assumed for the boron dilution event in the accident analyses. It is also consistent with the current STS. We, therefore, find the proposed changes to TS Pages 1-7, 3/4 1-16, 3/4 9-1 and B 3/4 9-1 acceptable as modified to agree with STS and accepted by the licensee.

Boration Requirements

The current TS action statements in Sections 3.1.1.1, 3.1.1.2, 3.10.1 and B 3/4.1.2 require "...immediately initiate and continue boration at \geq 40 gpm until the required SHUTDOWN MARGIN is restored". The proposed change would specify the source of, and thereby the minimum concentration of the boric acid solution.

Our review of this proposed TS change reveals the word "immediately" is used. About two years ago, we went through the exercise of replacing this word with a definite time interval. Discussions with NNECO have resulted in agreement to perform this action within 15 minutes of detecting a reduced shutdown margin.

We find the proposed changes to TS Pages 3/4 1-1, 3/4 1-3, 3/4 10-1 and B 3/4 1-3 are acceptable as modified to meet our requirements and agreed to by the licensee.

In addition, we suggested to NNECO that the maximum boron concentration of 2400 ppm used in the trisodium phosphate (TSP) calculation be identified in the TS basis. NNECO agreed that this would document the assumption used in this calculation. This change should be made.

Special Test Peak Linear Heat Rate

Amendment No. 61, (October 6, 1980) authorized low temperature operation for special tests during Cycle 4, only. To insure control of peaking factors, the allowable peak linear heat rate (LHR) was limited to 14.2 kW/ft instead of its normal value of 15.6 kW/ft for this type of operation. Request No. 9 is to remove this extra limit line from Figure 3.2.1. We find this change administrative in nature and, therefore, acceptable. The TS page involved is 3.4 2-3.

RTD Response Time Testing

In NRC letter dated June 30, 1981, we requested all CE designed units to propose TS to require resistance temperature detectors (RTDs) response time testing within one month for newly installed RTDs and once every eighteen months thereafter. Item 6 of NNECO's application proposes this requested change to TS 4.3.1.1. This change is needed to assure that RTDs used in the reactor protection system (RPS) do not experience degradation with time in use. We find this proposed increase in RTD response time testing acceptable. The affected page is 3/4 3-1.

Containment Purge Valves Actuation

Part of the No. 5 proposal of December 17, 1981 is to remove the Containment Isolation Actuation System (CIAS) and Safety Injection Actuation System (SIAS) auto closure of the containment purge valves from TS Tables 3.3-3 and 4.3-2. Previously, by Amendment No. 61 the TS were changed to require these valves to be locked closed in Modes 1-4. In

addition, we requested the electrical circuitry be physically disconnected. NNECO confirms that this has been done. Therefore, the TS Pages 3/4 3-14 and 3/4 3-24 should be modified to remove reference to the disconnected circuits.

ESFAS Allowable Values

The second part of the No. 5 proposal is to change the allowable values in Table 3.3-4 for the engineered safety features actuation system (ESFAS) trips. The proposed change is to specify more conservative allowable values than the current TS for containment pressure-high, pressurizer pressure-low, containment pressure-high-high, steam generator pressure-low and 4.16 kV emergency bus undervoltage-level one and level two. This change will make the ESFAS allowable values consistent with the RPS values. The allowable values include the maximum expected drift assumed to occur between surveillance intervals for each trip. In addition, the proposed change would remove the containment purge valve actuation (CIAS) and SIAS trip function requirements from Table 3.3-4 to be consistent with TS Table 3.3-3 and 4.3-2. We find these proposed changes to TS Pages 3/4 3-18, 3/4 3-19 and 3/4 3-20 acceptable based on the above comments.

Core Barrel Monitoring

NNECO proposal No. 4 requests that the core barrel monitoring requirements be deleted. The requirements of TS 3.4.11 and 4.4.11 for monitoring and reporting core barrel movement were originally required after the Palisades plant experienced excessive barrel motion. The vendor, Combustion Engineering (CE), devised a generic design modification which has been made to Millstone Unit No. 2. Considerable operating experience since has not revealed any excessive core barrel motion nor has inspection revealed excessive wear of the core barrel to reactor vessel interface nor in the flange area. Under those conditions we find that the proposed removal of the action statement for reporting requirements is acceptable. The affected pages are 3/4 4-23, 3/4 4-24 and B 3/4 4-12.

Ventilation System HEPA Filters

These proposed changes to TS 4.6.5.1, 4.7.6.1 and 4.9.15 surveillance requirements would bring these early edition TS on the enclosure building, control room and spent fuel pool ventilation systems into consistency with current TS. The NNECO proposal would bring the testing requirements on these systems into agreement with Regulatory Guide 1.52, Revision 2, and ANSI Standard N510-1975. These proposed TS are, therefore, acceptable. The affected pages are 3/4 6-25, 26 and 27, 3/4 7-16 17 and 18, and 3/4 9-16, 17 and 18.

Editorial Revisions

The December 17, 1981 application also contains several editorial changes to the following pages.

Page 3/4 3-4 - Replace "passed" with "bypassed" in table notation f.

Page 3/4 3-12 - Substitute Actions "1" and "2" of TS Table 3.3-3 for the current TS numbers "6" and "7" to be consistent with the table notation.

Page B 3/4 6-2 - The maximum peak containment pressure from the LOCA event should be increased from 51.2 to 53.8 psig. This results from the increased power approval for Cycle 3 and subsequent operation.

We find these editorial changes acceptable.

Environmental Consideration

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

Conclusion

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

Date: February 22, 1982

Principal Contributors:

Monte Conner
Rich Lobe
Felix Litton

UNITED STATES NUCLEAR REGULATORY COMMISSIONDOCKET NO. 50-336NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY
OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 72 to Facility Operating License No. DPR-65 issued to the Northeast Nuclear Energy Company, the Connecticut Light and Power Company, the Hartford Electric Light Company, and the Western Massachusetts Electric Company (the licensee), which revised Technical Specifications for operation of the Millstone Nuclear Power Station, Unit No. 2 (the facility) located in the Town of Waterford, Connecticut. The amendment is effective as of its date of issuance.

The amendment consists of changes to the Technical Specifications related to:

- Mode 6 reactivity control,
- Boration requirements,
- Special test peak linear heat rate,
- RTD response time testing,
- Containment purge valves actuation,
- ESFAS trip setpoints,
- Core barrel monitoring,
- Ventilation system HEPA filters, and
- Editorial revisions.

- 2 -

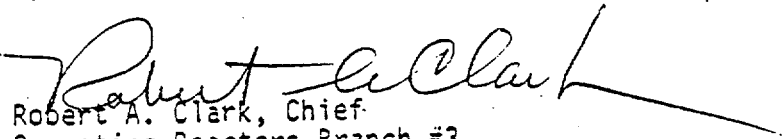
The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated December 17, 1981, (2) Amendment No. 72 to License No. DPR-65, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Waterford Public Library, Rope Ferry Road, Waterford, Connecticut. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 22nd day of February, 1982.

FOR THE NUCLEAR REGULATORY COMMISSION


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