~~JAN 14~~

JAN 14 1981

Docket No. 50-336

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

Dear Mr. Counsil:

The Commission has issued the enclosed Amendment No. 63 to Facility Operating License No. DPR-65 for Millstone Nuclear Power Station, Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your applications dated March 31, 1980, May 20, 1980 and August 29, 1980 as supplemented by letters dated November 28 and 30 and December 6 and 7, 1979 and January 17 and 25, March 5 and 10, April 11, June 16 and October 31, 1980.

The amendment:

- o adds operability, trip setpoint and surveillance requirements for automatic initiation of the auxiliary feedwater system;
- o increases the surveillance requirements on the auxiliary feedwater pumps and related flow paths.

Some portions of your proposed Technical Specifications have been modified to meet our requirements. These modifications have been discussed with and agreed to by your staff. The remaining Technical Specifications proposed by your letter of August 29, 1980 were addressed by Amendment No. 61 issued October 6, 1980.

The enclosed Safety Evaluation also documents our review of your responses to the staff's short-term and long-term recommendations that resulted from our reliability evaluation of the Millstone Unit No. 2 auxiliary feedwater system. These recommendations were the subject of our October 22, 1979 letter. Based on our review, we find your response to our recommendations acceptable with the exception of the following two items.

1. We do not consider the 27.5 hour endurance test of the "A" auxiliary feedwater pump sufficient to identify potential operating problems. You are requested to complete a 48 hour endurance test of this pump and to so document such a test within 30 days of receipt of this letter.

cp 1

OFFICE ▶							
SURNAME ▶							
DATE ▶	8101290799						

2. We continue to require automatic initiation of the steam turbine-driven auxiliary feedwater pump in order to provide reliable flow to the steam generators in the event of a station blackout. We consider this requirement a part of Item II.E.1.2.1.b, safety grade automatic initiation, and as such its required implementation date is July 1, 1981. You are requested to document your intentions regarding compliance with this requirement within 30 days of receipt of this letter.

The detailed review of the safety grade instrumentation system required to automatically initiate auxiliary feedwater system will be issued at a later time.

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

Sincerely,

Original signed by:

T. M. Novak, Assistant Director
For Operating Reactors
Division of Licensing

Enclosures:
1. Amendment No. 63 to DPR-65
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

*DENOTES PREVIOUS CONCURRENCE -
SEE ATTACHED

not required

		DSI:RSB*	DSI:I&CSB*	DSI:CBS *			
		TSpeis	RSatterfield	WButler			
		11/28/80	11/14/80	11/14/80			
OFFICE	DL:ORB#3 *	DL:ORB#3 *	DL:ORB#3:*	DL:AD:OR *	OELD *	STS	DSI:ASB
SURNAME	PMKretuzer	ELConner/JL	RAClark	TNNovak	JGray	JWetmore	ODParr
DATE	11/10/80	11/10/80 1/8/81	11/14/80	1/14/81	11/13/80	1/14/81	1/13/81

Original combined amendment.

HANDLED ON

DEC 23 1980

E. L. "MONTE" CONNER

Docket No. 50-336

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

Dear Mr. Counsil:

The Commission has issued the enclosed Amendment No. to Facility Operating License No. DPR-65 for Millstone Nuclear Power Station, Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your applications dated December 27, 1976, March 31, 1980, May 20, 1980 and August 29, 1980 as supplemented by letters dated July 27, 1977, January 16, November 28 and 30 and December 6 and 7, 1979 and January 17 and 25, March 5 and 10, April 11, June 16 and October 31, 1980.

The amendment:

- o adds operability, trip setpoint and surveillance requirements for automatic initiation of the auxiliary feedwater system;
- o allows credit for automatic testing of the engineered safety feature actuations system instrumentation; and
- o increases the surveillance requirements on the auxiliary feedwater pumps and related flow paths.

Some portions of your proposed Technical Specifications have been modified to meet our requirements. These modifications have been discussed with and agreed to by your staff. The remaining Technical Specifications proposed by your letter of August 29, 1980 were addressed by Amendment No. 61 issued October 6, 1980.

The detailed review of the safety grade instrumentation system required to automatically initiate auxiliary feedwater system will be issued at a later time.

Copies of the related Safety Ev...
enclosed...

RCW for T&D 11/14/80

DSI:ASB	DSI:RSB	DSI:I&CSB	DSI:CSB
ODParr	TSpeis	RSatterfield	WButler
11/ /80	11/20/80	11/ /80	11/14/80

Amend & FR Notice only - note error on p. 3/4 7-4 of Tech Spec change

OFFICE	DL:ORB#3	DL:ORB#3	DL:ORB#3	DL:AD:OR	OELD	STSG
SURNAME	PKreutzer:jl	ELConner	RAClark	TMNovak	JGray	JWetmore
DATE	11/16/80	11/10/80	11/14/80	11/14/80	11/13/80	11/ /80

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JAN 27 1981
U.S. NUCLEAR REGULATORY COMMISSION

OFFICE							
SURNAME							
DATE							

Northeast Nuclear Energy Company

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

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N.W.
Washington, D.C. 20005

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

Director, Technical Assessment Division
Office of Radiation Programs (AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection Agency
Region I Office
ATTN: EIS COORDINATOR
John F. Kennedy Federal Building
Boston, Massachusetts 02203

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

Northeast Utilities Service Company
ATTN: Mr. James R. Himmelwright
Nuclear Engineering and Operations
P. O. Box 270
Hartford, Connecticut 06101

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

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

cc w/enclosure(s) and incoming
dtd.: 3/31/80; 5/20/80 & 8/29/80
Connecticut Energy Agency
ATTN: Assistant Director, Research and Policy Development
Department of Planning and Energy Policy
20 Grand Street
Hartford, Connecticut 06106

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

January 15, 1981

DISTRIBUTION:
Docket File
ORB#3 Rdg
PMKreutzer

Docket No. 50-336

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: **MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2**

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

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

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

☐ Notice of Availability of Applicant's Environmental Report.

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

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

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

☐ Notice of Limited Work Authorization.

☐ Notice of Availability of Safety Evaluation Report.

☐ Notice of Issuance of Construction Permit(s).

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

☒ Other: Amendment No. 63
Referenced documents have been provided PDR.

Division of Licensing, ORB#3
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE	ORB#3:DL					
SURNAME	PMKreutzer/pn					
DATE	1/15/81					

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. 63
License No. DPR-65

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

A. The applications for amendment by Northeast Nuclear Energy Company (the licensee) dated March 31, May 20 and August 29, 1980, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

B. The facility will operate in conformity with the 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.

8101290802

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. 63 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective on January 20, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 14, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 63

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 indicated the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

XVII
3/4 3-13
3/4 3-14
3/4 3-15
3/4 3-16
3/4 3-17
3/4 3-20
3/4 3-21
3/4 3-22
3/4 3-24
3/4 7-4
3/4 7-5
B 3/4 7-2

ADMINISTRATIVE CONTROLS

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
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

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. CONTAINMENT PURGE VALVE ISOLATION					
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. Automatic CIAS Actuation Logic	2	1	2	1, 2, 3	1
d. Containment Radiation - High					
Gaseous Monitor	1(d)	1(d)	1	1, 2, 3, 4, 6	3
Particulate Monitor	1(d)	1(d)	1	1, 2, 3, 4, 6	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

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MIMIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. AUXILIARY FEEDWATER					
a. Manual	1/pump	1/pump	1/pump	1, 2, 3	1
b. Steam Generator Level - Low	4	2	3	1, 2, 3	2

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed when pressurizer pressure is < 1750 psia; bypass shall be automatically removed when pressurizer pressure is ≥ 1750 psia.

(b) An SIAS signal is first necessary to enable CSAS logic.

(c) Trip function may be bypassed below 600 psia; bypass shall be automatically removed at or above 600 psia.

(d) Each channel has two sensors, high radiation level on either sensor will initiate containment purge valve isolation.

(e) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in COLD SHUTDOWN within the next 36 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels and with the pressurizer pressure:

a. < 1750 psia; immediately place the inoperable channel in the bypassed condition; restore the inoperable channel to OPERABLE status prior to increasing the pressurizer pressure above 1750 psia.

b. ≥ 1750 psia, operation may continue with the inoperable channel in the bypassed condition, provided the following conditions are satisfied:

1. All functional units receiving an input from the bypassed channel are also placed in the bypassed condition.

2. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be removed from service for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided one of the inoperable channels is placed in the tripped condition.

TABLE 3.3-3 (Continued)

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

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	$\leq$ 5 psig	$\leq$ 5 psig
c. Pressurizer Pressure - Low	$\geq$ 1600 psia	$\geq$ 1600 psia
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High-High	$\leq$ 27 psig	$\leq$ 27 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	$\leq$ 5 psig	$\leq$ 5 psig
d. Pressurizer Pressure - Low	$\geq$ 1600 psia	$\geq$ 1600 psia
4. MAIN STEAM LINE ISOLATION		
a. Steam Generator Pressure - Low	$\geq$ 500 psia	$\geq$ 500 psia

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
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	≤ 5 psig	≤ 5 psig
d. Pressurizer Pressure - Low	≥ 1600 psia	≥ 1600 psia
6. CONTAINMENT SUMP RECIRCULATION (SRAS)		
a. Manual SRAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Storage Tank - Low	48 ± 18 inches above tank bottom	48 ± 18 inches above tank bottom
7. CONTAINMENT PURGE VALVES ISOLATION		
a. Manual CIAS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Automatic CIAS Actuation Logic	Not Applicable	Not Applicable
d. 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

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

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. SIAS	
Safety Injection (ECCS)	Not Applicable
Containment Isolation	Not Applicable
Enclosure Building Filtration System	Not Applicable
b. CSAS	
Containment Spray	Not Applicable
c. CIAS	
Containment Isolation	Not Applicable
d. SRAS	
Containment Sump Recirculation	Not Applicable
e. EBFAS	
Enclosure Building Filtration System	Not Applicable
f. Auxiliary Feedwater Initiation	Not Applicable
2. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	
1) High Pressure Safety Injection	≤ 30.0*/5.0**
2) Low Pressure Safety Injection	≤ 50.0*/5.0**
3) Charging Pumps	≤ 40.0*/40.0**
4) Containment Air Recirculation System	≤ 31.0*/31.0**
b. Containment Isolation	≤ 7.5
c. Enclosure Building Filtration System	≤ 50.0*/50.0**

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Containment Pressure-High	
a. Safety Injection (ECCS)	
1) High Pressure Safety Injection	≤ 30.0*/5.0**
2) Low Pressure Safety Injection	≤ 50.0*/5.0**
3) Charging Pumps	≤ 40.0*/40.0**
4) Containment Air Recirculation System	≤ 31.0*/31.0**
b. Containment Isolation	≤ 7.5
c. Enclosure Building Filtration System	≤ 50.0*/50.0**
4. Containment Pressure--High-High	
a. Containment Spray	≤ 35.6*(1)/35.6**(1)
5. Containment Radiation-High	
a. Containment Purge Valves Isolation	≤ Counting period plus 7.5
6. Steam Generator Pressure-Low	
a. Main Steam Isolation	≤ 6.9
b. Feedwater Isolation	≤ 60
7. Refueling Water Storage Tank-Low	
a. Containment Sump Recirculation	≤ 120
8. Steam Generator Level-Low	
a. Auxiliary Feedwater System	≤ 240*/240**(2)

TABLE NOTATION

* Diesel generator starting and sequence loading delays included.

** Diesel generator starting and sequence loading delays not included. Offsite power available.

(1) Header fill time not included.

(2) Includes 3-minute time delay.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
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

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
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. Manual CIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
c. Automatic CIAS Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
d. 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

TABLE 4.7-1

STEAM LINE SAFETY VALVES

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

AUXILIARY FEEDWATER PUMPS

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one auxiliary feedwater pump inoperable, restore at least three auxiliary feedwater pumps (two capable of being powered from separate OPERABLE emergency busses and one capable of being powered by an OPERABLE steam supply system) to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Starting each pump from the control room,

2. Verifying that:

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

b) The steam turbine driven pump develops a discharge pressure of $\geq$ 1080 psig on recirculation flow.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that each pump operates for at least 15 minutes.

4. Cycling each testable, remote operated valve through at least one complete cycle.

5. Verifying the correct position for each manual valve not locked, sealed or otherwise secured in position.

6. Verifying the correct position for each remote operated valve.

b. Before entering MODE 3 after a COLD SHUTDOWN of at least 30 days by completing a flow test that verifies the flow path from the condensate storage tank to the steam generators.

c. At least once per 18 months during shutdown by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of each auxiliary feedwater actuation test signal.

2. Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of each auxiliary feedwater actuation test signal.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank shall be OPERABLE with a minimum contained volume of 150,000 gallons.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With less than 150,000 gallons of water in the condensate storage tank, within 4 hours either:

a. Restore the water volume to within the limit or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of the fire water system as a backup supply to the auxiliary feedwater pumps and restore the condensate storage tank water volume to within its limits within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the water level.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1100 psig) of its design pressure of 1000 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 12.7×10^6 lbs/hr which is 108 percent of the total secondary steam flow of 11.8×10^6 lbs/hr at 100% RATED THERMAL POWER.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip setpoint reductions are derived on the following bases:

For two loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times 106.6$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

106.6 = Power Level-High Trip Setpoint for two loop operation

X = Total relieving capacity of all safety valves per steam line in lbs/hour = 6.35×10^6 lbs/hour

Y = Maximum relieving capacity of any one safety valve in lbs/hour = 7.94×10^5 lbs/hour

3/4.7.1.2 AUXILIARY FEEDWATER PUMPS

The OPERABILITY of the auxiliary feedwater pumps ensures that the Reactor Coolant System can be cooled down to less than 300°F from normal operating conditions in the event of a total loss of off-site power.

Any single motor driven or steam driven pump has the required capacity to provide sufficient feedwater flow to remove reactor decay heat and reduce the RCS temperature to $\leq$ 300°F where the shutdown cooling system may be placed into operation for continued cooldown.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available for cooldown of the Reactor Coolant System to less than 300°F in the event of a total loss of off-site power. The minimum water volume is sufficient to maintain the RCS at HOT STANDBY conditions for 10 hours with steam discharge to atmosphere.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 63 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

Early in the review of the Three Mile Island Unit No. 2 (TMI-2) accident, it became apparent that increased plant safety would result from automatic initiation of auxiliary feedwater system (AFWS) flow. This was short-term recommendation No. 2.1.7a of our July 1979 NUREG-0578. In the implementation letters dated September 13 and October 30, 1979, we provided clarification of requirement No. 2.1.7a and proposed control grade system installation by January 1, 1980 with the upgrading of the automatic initiation of AFWS flow to safety grade by January 1, 1981.

In letters dated November 21 and 30, 1979, Northeast Nuclear Energy Company (NNECO) pointed out that modifying the AFWS to be automatically initiated was not necessary and constituted an unreviewed safety question issue since AFWS flow was not considered in the Millstone Unit No. 2 (Millstone-2) main steam-line break (MSLB) analyses. NNECO (and other licensees) contend that the addition of AFW flow during a MSLB accident will: (1) result in a positive reactivity insertion (due to increased cooldown) and, thus, a higher final return-to-power condition; and (2) a higher peak containment pressure than the values calculated in the analysis of record. Reiterating their concern about the unreviewed safety question, NNECO proposed, by letters dated December 6 and 17, 1979, their control design for automatic initiation of AFWS flow.

Our letters of December 21 and 27, 1979 address the NNECO concern. We agreed that AFWS flow may adversely affect the MSLB accident and requested a re-analysis of this accident to be submitted for our review prior to the final connection of the circuits involved to automatically initiate AFWS flow. The requested reanalysis was supplied by the NNECO letter of January 25, 1980 as supplemented by letter of April 11, 1980. This Safety Evaluation (SE) will review the effects of automatic initiation of AFWS flow on the likelihood of return to power (SE Section 2.1) and on the calculated peak containment pressure (SE Section 2.2) during the main steamline break accident. The adverse effects of delaying AFWS flow for three minutes on other transients and accidents will be addressed in SE Section 2.3.

In the application for Technical Specifications (TS) changes for automatic initiation of the AFWS, NNECO provided an analysis redefining the pump rated capacity. This analysis will be evaluated in Section 2.4 of this SE. The proposed TS changes for the AFWS applications dated March 31, May 20 and August 29, 1980 will be addressed in Section 2.6.

Our letter of October 22, 1979 presented the staff reliability evaluation of the Millstone-2 AFWS and made short-term and long-term recommendations. NNECO's responses to these recommendations were submitted by letters dated November 28, 1979 and January 17, March 10, and June 16, 1980. Our review of this information is presented in Section 2.5 of this SE.

The detailed review of the safety grade instrumentation system required to automatically initiate the AFWS is to be reviewed by the Franklin Research Center in Philadelphia, Pennsylvania, under NRC contract. The resultant Safety Evaluation will be issued at a later time.

2.0 Discussion and Evaluation

2.1 MSLB Accident - Return to Power

NNECO's analysis of the effects of return to power following a MSLB accident is presented in Attachment 1 to their January 25, 1980 letter. The starting conservative assumptions, according to NNECO, for this analysis are:

- Only a three-minute delay in delivery of auxiliary feedwater flow to the steam generators was assumed, rather than a more realistic longer time delay,
- Credit is not taken for complete isolation of the main feedwater system, thereby resulting in a continuous flow of 772 gpm (5 percent of full flow) of main feedwater to the affected steam generator,
- A conservative representation of auxiliary pump feedwater flow, namely 2800 gpm, which is 35% higher than maximum runout flow at Millstone Unit No. 2. Thus, a total of 3572 gpm of feedwater flow is assumed in the analysis,
- Failure of one HPSI pump,
- Failure of one LPSI pump,
- The highest worth CEA is assumed to stick in the fully withdrawn position, and
- The end of Cycle 3 moderator temperature and Doppler (fuel temperature) coefficient values were used since these values result in the greatest positive reactivity change during cooldown.

The analysis assumed that the event is initiated by a circumferential rupture of a 34 inch main steam line at the steam generator nozzle. NNECO states that this break is limiting since it results in the greatest rate of temperature reduction in the reactor core region. The reanalysis reported in the January 25, 1980 submittal uses the same assumptions and methods as previously used except that it simulates automatic initiation of auxiliary feedwater flow in three minutes from initiation of the event.

The rationale for delaying the initiation of AFWS originates from the positive reactivity feedback which accompanies a postulated MSLB. During a postulated double-ended guillotine break of this steam line, the broken steam generator behaves as an enhanced heat sink, resulting in rapid cooldown of the primary system. This rapid cooldown has a noticeable impact on the moderator reactivity feedback, which results in a net positive reactivity insertion. A conservative assumption is made that the limiting control element assembly (CEA) is stuck in its fully withdrawn position.

Based on the licensee's generic analyses, the reactivity feedback was most limiting during full power operation. For this condition, the calculations predict that there would be a 10.8% return to power due to the cooling effect of the auxiliary feedwater; however, this is less than the 12% return to power predicted prior-to-auxiliary feedwater injection. The net energy removed from the primary system was conservatively assumed to be the product of the total steam generator secondary mass (M_{TOT}) times the latent heat of evaporation (h_{fg}). Should liquid entrainment exit the break, then the energy removed from the primary system will be less severe.

For a postulated guillotine break in a steam line, the time required to deplete the broken steam generator secondary inventory is approximately 1 1/2 minutes (for the full power condition). Should the auxiliary feedwater inject into the steam generator immediately, when called upon to do so, then the magnitude of the primary side cooldown is increased (M_{TOT} x h_{fg}; where M_{TOT} is increased). This results in enhancing the primary side cooling and in an increased reactivity feedback. The mechanism available for turning the reactivity around is the initiation of ECCS, which injects boron into the system.

Due to the time constraints in providing analytical assessment of auto-initiation of the AFWS, a generic review was conducted by Combustion Engineering (CE). This review concluded that a three-minute delay in the initiation of AFWS will ensure that the DNBR limit will not be exceeded. Fuel rods which exceed the DNBR limit are assumed to fail (a conservative assumption).

The purpose of the three-minute delay is to provide time for the ECCS injected borated water to lessen the magnitude of the moderator reactivity feedback attributed to the AFWS inventory. Analyses have shown that during full power operation, the core becomes critical during the blowdown of a steam generator. After the steam generator has blown dry, the auxiliary feedwater injects, thus creating a second return to criticality, but at a magnitude less than experienced during the blowdown phase.

The licensee's analytical method for analyzing steam line breaks is presently under staff review. The review at this time indicates resaonable assurance that the conclusions based on the submitted analyses will not be appreciably altered by the completion of the analytical methods review. The staff finds the return to power results following a MSLB accident with automatically initiated AFWS flow delayed three minutes are not more limiting than previous analysis results without automatic AFWS flow and are, therefore, acceptable.

NNECO states that single failures concurrent with the MSLB, other than those listed in the assumptions, as well as loss of offsite power concurrent with MSLB, are not and have not been part of the design basis as described in the FSAR and, therefore, were not considered.
other significant failure would be one that results in a higher feedwater flow that was assumed in the present analysis. The present analysis assumed that main feedwater flow was reduced from 100 to 5 percent in 60 seconds by the feedwater control system. In reality, the feedwater flow would be reduced from 100 to 0 percent in less than 10 seconds by the action of two safety grade systems. These systems, the main feedwater isolation and the main steam isolation valves, are actuated by low steam generator pressure and cause the closure of the main feedwater isolation valves and main feedwater pump trip. By not taking credit for these systems, the present analysis contains a multiple failure assumption. From our review, we conclude that single failure has been treated in an acceptable manner in this analysis for Millstone 2.

The primary consequences resulting from loss-of-offsite power (LOOP) are a delay of emergency core cooling systems (ECCS) injection and tripping of the reactor coolant pumps. During LOOP, ECCS injection is delayed approximately 25 seconds as the emergency diesel generators restore power to the ECCS pumps. LOOP also results in coastdown of the reactor coolant pumps.

Continued operation of the reactor coolant pumps would have two effects on an SLB transient:

- Running the reactor coolant pumps (RCPs) results in a greater degree of overcooling as the hot primary fluid is forced through the steam generators, and
- The reactor coolant pumps act as a driving head, forcing the ECCS injected borated water into the core.

Thus, losing offsite power affects the degree of system cooling and the time at which the ECCS-injected boron enters the reactor core. Overcooling and borated water injection are competing effects in which the former increases reactivity and the latter reduces reactivity. In reviewing past analyses of MSLB for other plants similar to Millstone-2, we have determined that reduction in the RCS cooldown rate caused by coastdown of the RCP after LOOP has had a larger effect than slower boron injection to the core. Thus, we find that the MSLB accident is reduced in severity with a concurrent loss of offsite power.

We find automatic initiation of the auxiliary feedwater system to inject needed makeup water to the steam generators without the need for operator action will improve the nuclear safety of Millstone-2. The staff plans to perform independent audit calculations by the end of FY 81 to provide further confirmation of our conclusions.

2.2 MSLB Accident - Peak Containment Pressure

Appendix B of NNECO's January 25, 1980 letter provides a response to questions posed by our letter of December 21, 1979. Specifically, NNECO was to assess the potential for containment overpressurization due to the anticipated continuous addition, at pump runout flow, of auxiliary feedwater to the affected steam generator following a postulated MSLB accident. Automating the auxiliary feedwater system would cause an increase in energy released to containment after a MSLB thereby increasing the containment pressure.

The original FSAR analysis of the MSLB accident was based on the no load, single loop nozzel break case with a 20% moisture content in the blowdown. The results of this analysis were a peak containment pressure of 47 psig and a peak temperature of 279°F. NNECO states that no AFWS flow was assumed in the original analysis based on operating procedures which require isolation of the affected steam generator prior to manual AFWS initiation.

In NNECO's reanalysis, two containment pressure calculations were performed to envelope the effects of a single active failure upon the containment pressure response. The most limiting single failure resulted from assuming failure of one diesel generator with the resultant loss of one-half of the important ESF (one containment spray pump and two containment air recirculation fans). The AFWS pump run-out flow used was 2050 gpm, based on a conservative backpressure of one atmosphere. This reanalysis shows that the peak containment pressure remains 47 psig if AFWS flow is delayed for three minutes. It assumes the affected steam generator is not isolated resulting in a second increase of containment pressure up to almost 45 psig.

The staff concurs with the licensee's conclusion that the peak containment pressure will remain below the containment design pressure after the MSLB accident with the addition of auxiliary feedwater at the run-out flow rate three minutes after low steam generator level is reached.

Our review also included evaluation of the licensee's ability to determine and isolate the affected steam generator. NNECO states that the key parameter available to the operator following an MSLB would be low steam generator pressure in the affected steam generator. The MSLB analysis indicates automatic MSIV closure initiated at approximately three seconds after the break and a secondary side pressure of 500 psia (trip setpoint) in the affected steam generator versus approximately 695 psia in the intact steam generator. The mismatch becomes greater, approximately 98 psia in the affected steam generator versus 547 psia in the intact steam generator at 80 seconds after the break. The plant operating procedures are written to enable a quick determination of the steam line rupture and affected steam generator. Once the determination is completed approximately five seconds are required to manually isolate the affected steam generator stopping AFWS flow.

Based upon the above described control room indications, we find sufficient justification to assume the operator will be alerted to the need to isolate the AFWS flow path to the affected steam generator before initiating AFWS flow manually or within 10 minutes if automatic initiation is relied upon.

2.3 Effects of Three Minute Delay of AFWS Flow on Other Transients and Accidents

In addition to reviewing the effects of automatically initiating the AFWS in three minutes on the MSLB accident, we considered any adverse effects upon other transients and accidents. For example, assuming liquid discharge from a ruptured feedwater line, the reactor would lose one steam generator as a heat sink. A delay of AFWS injection could extend the heatup of the primary coolant system; however, the intact steam generator requires in excess of 10 minutes to boil dry and, therefore, provides an adequate heat sink for decay power removal.

Millstone's Operating Procedures have historically required the initiation of AFWS as a manual action. Whenever credit for operator action was required, the analysis performed demonstrated the acceptability of the unit to withstand the postulated event being independent of operator action for a minimum of 10 minutes. We, therefore, conclude that automatic initiation of AFWS flow three minutes into the transient or accident (versus 10 minutes assuming operator action) is appropriate and would not result in consequences more limiting than previously analyzed.

2.4 AFWS Pump Rated Capacity

The Millstone-2 AFWS contains two electric motor-driven pumps, one supplied emergency power from each diesel generator, and one steam turbine-driven pump with twice the rated capacity of either electric-driven pump. The design objectives of the system, according to FSAR Section 10.4.5.3 is to provide feedwater for the removal of sensible and decay heat and to cool the primary system to 300°F in case the main condenser and steam generator feed pumps are inoperative due to loss of normal electric power sources or the main steam. The steam turbine-driven pump with a capacity of 600 gpm at 2437 foot head was considered a "full-capacity" pump. The electric-driven pumps with capacities of 300 gpm each at 2437 foot head were "half-capacity" rated.

By application dated May 20, 1980, NNECO provided a reanalysis to show that the capacity of one electric motor-driven pump (300 gpm) was adequate to meet the sensible and decay heat requirements to cool the RCS. This was done to justify automatically starting only the two electric motor-driven pumps. The assumptions used were:

- One of the electric motor-driven pumps is inoperable;
- The turbine bypass to the condenser is unavailable;
- The AFWS pump starts 240 seconds after the steam generator low level is reached.

This reanalysis shows the steam generator inventory decreases to a minimum of 9,400 lbs per generator, about 70 minutes from the reactor trip, before recovery starts. The steam generator inventory loss and recovery can be improved by manually starting the steam turbine-driven pump. The reanalysis further indicates the peak RCS pressure occurs about nine minutes into the event and that the PORVs (setpoint 2400 psi) will not be automatically lifted.

Our criteria has been "to automatically initiate AFWS flow". Following the criteria for other ESF systems, however, we agree with NNECO that installing a circuit to automatically initiate two independently powered electric motor-driven AFWS pumps, each rated at 100% capacity, is adequate to meet our short term requirements.

NNECO finds some disadvantages to automatically initiating all three pumps. First, they contend that normal unit startup and shutdown is routinely made with only one electric motor-driven pump. Starting all three pumps at one time automatically would provide 400% of needed flow. Under these conditions of excess AFWS flow, NNECO believes it would be difficult to control the RCS cooldown rate within the limit specified in the TS.

Secondly, NNECO has noted cavitation of the Millstone-2 AFWS pumps at excessive flow rates and states that starting all three pumps simultaneously could result in possible impeller damage.

The final disadvantage was addressed in NNECO's letter of January 25, 1980. They state that it was recognized that during a postulated steam generator tube rupture event, steam generator levels will drop to the automatic initiation setpoint of the auxiliary feedwater system, which would include the Terry turbine. This turbine exhausts directly to atmosphere. Although use of the Terry turbine was previously a possibility during a steam generator tube rupture event, the modification under consideration would result in an automatic start of the Terry turbine during this event. Therefore, NNECO performed a reanalysis of the radiological consequences of this event using the methodology previously documented, and assuming a release for thirty minutes from the Terry turbine exhaust. The results of this analysis are:

	Thyroid Dose (Rem)	Whole Body Dose (Rem)
• Results presented in Reference Cycle Analysis	0.006	0.1
• Additional Dose From Terry Turbine	0.048	0.003
• Total	0.054	0.1

We find that although the radiation exposure increase due to running the turbine-driven pump is not a significant increase from past analysis and is a small fraction of the dose guidelines of 10 CFR 100, added to the possibility of RCS overcooling and pump cavitation, there is a need to carefully consider these factors in the design of the automatic start circuit for the turbine-driven pump.

We conclude that automatic initiation of the two electric motor-driven AFWS pumps meets our short term requirements. However, automatic initiation of the steam turbine driven AFWS pump is necessary to meet our October 22, 1979 requirement GL-3. See Section 2.5.3 of this SE for an evaluation of GL-3.

2.5 NNECO Response to NRC Recommendations

In response to our letter of October 22, 1979, NNECO provided responses to our short-term and long-term recommendations by letters dated November 28, 1979 and January 17, March 10 and June 16, 1980. Our evaluation of these responses is as follows:

2.5.1 Short Term Recommendations

Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following should be covered by the procedures:

- The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

- The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

Evaluation GS-4 - The licensee's response is acceptable in that applicable emergency procedures were revised to reflect the requirement for transfer to alternate sources of AFW supply before January 1, 1980.

Recommendation GS-5 - The plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

Evaluation GS-5 - The licensee's response is acceptable. They have confirmed that emergency procedures were revised before January 1, 1980 to contain the information required to provide AFWS flow from one pump, independent of AC power source, for at least two hours. Adequate portable lighting and communication are available for the prescribed manual actions.

Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

- Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

- The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

Evaluation GS-6 - The licensee's response is acceptable. It was our position that procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned. The licensee has agreed to perform this valve verification in accordance with our position. By letter dated March 31, 1980, the licensee proposed modifications to the TS to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The AFWS valves will be in their normal alignment during the flow test.

Recommendation GS-8 - The licensee should install a system to automatically initiate AFW system flow. For the short-term, this system need not be safety-grade; however, it should meet the criteria listed below, which are similar to Item 2.1.7a of NUREG-0578. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GS-1.

- The design should provide for the automatic initiation of the auxiliary feedwater system flow.
- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
- Testability of the initiating signals and circuits should be a feature of the design.
- The initiating signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits should be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

Evaluation GS-8 - The licensee's response is acceptable. The NNECO letter of May 20, 1980 proposed to automatically start the two electrically driven AFW pumps and hold the turbine driven AFW pump as a manual backup. Their analysis shows that one electrically driven AFW pump (assuming failure of the second pump) will keep the steam generator level above the tubes sufficient to remove decay heat from the reactor core. See Section 2.4 of this SE.

2.5.2 Additional Short-Term Recommendations

Recommendation 1 - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFWS pump is operating.

Evaluation 1 - The licensee's response is acceptable as recently modified. The installed level instrumentation in the condensate storage tank consists of low level alarm and a low-low level alarm in the control room. The licensee has confirmed that both the low and low-low level set points on the condensate storage tank are set to allow more than 20 minutes for operator action assuming that the largest capacity AFWS pump is operating.

Recommendation 2 - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

Evaluation 2 - The staff has reduced the required performance test to 48 hours of operation instead of 72 hours. The NNECO response is acceptable for the "B" motor driven and the steam turbine driven AFW pumps. Data provided in the NNECO letter of January 17, 1980 documents reliable operation of these two AFWS pumps for considerable length of time. The licensee should be required to conduct a 48-hour endurance test of the "A" motor driven AFW pump within 30 days from issuance of the SE.

Recommendation 3 - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

- Safety-grade indication of auxiliary feedwater flow to each steam generator should be provided in the control room.

- The auxiliary feedwater flow instrument channels should be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

Evaluation 3 - The licensee's response is acceptable. NNECO has committed to provide safety grade AFWS flow indicator to each steam generator by July 1, 1981, as required by NUREG 0737.

Recommendation 4 - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFWS train, and there is only one remaining AFWS train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFWS train from the test mode to its operational alignment.

Evaluation 4 - NNECO's response is acceptable. The licensee indicates that local manual valve realignment is not required for periodic testing.

2.5.3 Long-Term Recommendations

Recommendation GL-1 - Licensees with plants having a normal starting AFWS should install a system to automatically initiate the AFWS flow. This system and associated automatic initiation signals should be designed and installed to meet safety-grade requirements. Manual AFWS start and control capability should be retained with manual start serving as backup to automatic AFWS initiation.

Evaluation GL-1 - NNECO has installed the control grade circuitry required to automatically start the two electric motor-driven AFWS pumps. They have also committed to upgrade and replace components as necessary to meet safety-grade requirements. Our review of the safety-grade components will be completed and issued at a later date.

Recommendation GL-3 - At least one AFW system pump and its associated flow path and essential instrumentaion should automatically initiate AFWS flow and be capable of being independent of any alternating current power source for at least two hours. Conversion of direct power to alternating current is acceptable.

Evaluation GL-3 - NNECO has stated that the upgrading of the turbine driven AFWS, by removing all AC dependence, will be completed by January 1, 1982. We find this will improve the reliability of the turbine driven AFWS during a station blackout. However, during such a postulated event, the control room personnel could be under considerable pressure to restore an electrical source thereby diverting their attention from the manual initiation of AFWS flow by starting the steam turbine-driven pump. Therefore, we continue to require automatic initiation of the steam turbine-driven AFWS pump in order to meet the criteria of specific Requirement GL-3 of our October 22, 1979 letter. This criteria does not preclude a design with special features to alleviate the disadvantages that NNECO identified in their letters of January 25 and May 20, 1980, as discussed in section 2.3 of this SE.

2.6 Technical Specifications Changes

The proposed TS changes under review are from NNECO's applications dated December 27, 1976 and March 31, May 20 and August 29, 1980. Some portions of the proposed TS changes or related TS pages should be modified to meet our requirements or for increased clarification. Such modifications have been discussed with and agreed to by the NNECO staff.

Pages 3/4 3-13, 3/4 3-14, 3/4 3-16 and 3/4 3-17

Editoral change to correct numbers of action items.

Pages 3/4 3-15, 3/4 3-20, 3/4 3-21 and 3/4 3/22

The automatic initiation of AFWS requirements should be added to the ESFAS Tables 3.3-3, 3.3-4 and 3.3-5.

Page 3/4 3-24

The surveillance requirements for automatic initiation of AFWS should be added to Table 4.3-2.

Pages 3/4 7-4 and 3/4 7-5

The surveillance requirements needed to prove the operability of the AFWS should be expanded to include automatic initiation, flow path verification and valve alignment.

Page B 3/4 7-2

The AFWS pump basis should be changed to specify the new capacity rating of the electric or turbine-driven pumps.

3.0 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.

4.0 Safety 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: January 14, 1981

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-336

NORTHEAST 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. 63 to Facility Operating License No. DPR-65 issued to the Northest 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 on January 20, 1981.

The amendment adds operability trip setpoint and surveillance requirements for automatic initiation of the auxiliary feedwater system and increases the surveillance requirements on the auxiliary feedwater pumps and related flow paths.

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.

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

For further details with respect to this action, see (1) the applications for amendment dated March 31, May 20 and August 29, 1980, (2) Amendment No. 63 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 14th day of January, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

R. A. Clark

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