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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. 66 to Facility Operating License No. DPR-65 for Millstone Nuclear Power Station, Unit No. 2. This amendment consists of changes to the Technical Specifications (TS) in response to your application dated September 16, 1980.

The changes to the TS incorporate certain of the Lessons Learned Category "A" requirements related to the Three Mile Island Accident in direct response to our request dated July 2, 1980.

Certain modifications to your proposed changes were necessary to meet our criteria. These modifications have been discussed with and agreed to by your staff.

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

Sincerely,

Original signed by
Robert A. Clark

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

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Enclosures:
1. Amendment No.66 to DPR-65
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

Concur in license amendment and Fed. Reg. notices
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DATE	4/7/81	3/5/81	3/ /81	3/6/81	3/9/81	3/16/81	

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UNITED STATES
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WASHINGTON, D.C. 20555

DISTRIBUTION:
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PMKreutzer

Docket No. 50-336

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: NORTHEAST NUCLEAR ENERGY COMPANY, ET AL., 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).

X☒ Other: Amendment No. 66.
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/pr					
DATE	4/10/81					

NRC FORM 102 7—79

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

April 7, 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:

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The changes to the TS incorporate certain of the Lessons Learned Category "A" requirements related to the Three Mile Island Accident in direct response to our request dated July 2, 1980.

Certain modifications to your proposed changes were necessary to meet our criteria. These modifications have been discussed with and agreed to by your staff.

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

Sincerely,

Robert A. Clark

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

Enclosures:
1. Amendment No. 66 to DPR-65
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

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

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

Director, Criteria and Standards Division
Office of Radiation Programs (ANR-460)
U.S. Environmental Protection Agency
Washington, D. C. 20460

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

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.N.R.C.
P. O. Drawer KK
Niantic, CT 06357

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

cc w/enclosure(s) and incoming
dtd: 9/16/80

Connecticut Energy Agency
ATTN: Assistant Director, Research
and Policy Development
Department of Planning and Energy
Policy
20 Grand Street
Hartford, CT 06106

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. 66
License No. DPR-65

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

   A. The application for amendment by Northeast Nuclear Energy Company (the licensee) dated December 27, 1976, 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.

8104160060

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. 66 , 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 7, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 66

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

IV
X
XVII
3/4 3-46
3/4 3-47
3/4 3-48
3/4 3-49
3/4 4-2
3/4 4-3
3/4 4-4
B 3/4 3-4
B 3/4 4-1
B 3/4 4-2
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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

BASES

SECTION PAGE

INDEX

ADMINISTRATIVE CONTROLS

TABLE 3.3-10 (Continued)

FIRE DETECTION INSTRUMENTS

Instrument Location (Zone)	Heat: Total No. of Channels	Heat: Minimum Channels Operable	Smoke: Total No. of Channels	Smoke: Minimum Channels Operable
5. Battery Rooms				
West Battery Room (14'6") (39)	--	--	1	1
East Battery Room (14'6") (39)	--	--	2	1
6. Electrical Penetration Rooms				
East (14'6") (20)	--	--	3	2
West (14'6") (17)	--	--	2	1
7. Diesel Generators				
Diesel 1221 (30)	--	--	1	1
Diesel 1321 (32)	--	--	1	1
8. Main Exhaust Equipment Room				
Room (EL 38'6") (5)	--	--	2	1

ACCIDENT MONITORING

LIMITING CONDITION FOR OPERATION

3.3.3.8 The accident monitoring instrumentation channels shown in Table 3.3.11 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. Actions per Table 3.3-11.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
1. Pressurizer Water Level	2	1	1
2. Auxiliary Feedwater Flow Rate	1/S. G.	1/S. G.	1
3. RCS Subcooling Margin Monitor	1	1	2
4. PORV Position Indicator Acoustic Flow Monitor	1/valve	1/valve	3
5. PORV Block Valve Position Indicator	1/valve	1/valve	3
6. Safety Valve Position Indicator Acoustic Flow Monitor	1/valve	1/valve	3

TABLE 3.3-11 (Continued)

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels less than required by Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 30 days or be in HOT STANDBY within the next 12 hours.

ACTION 2 - With the subcooling margin monitor INOPERABLE, determine the subcooling margin once per 12 hours.

ACTION 3 - With any individual valve position indicator inoperable, obtain quench tank temperature, level and pressure information, and monitor discharge pipe temperature once per shift to determine valve position.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Pressurizer Water Level	M	R
2. Auxiliary Feedwater Flow Rate	M	R
3. Reactor Coolant System Subcooling Margin Monitor	M	R
4. PORV Position Indicator	M	R
5. PORV Block Valve Position Indicator	M	R
6. Safety Valve Position Indicator	M	R

3/4.4 REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1 Four reactor coolant pumps shall be in operation.

APPLICABILITY: As noted below, but excluding MODE 6.*

ACTION:

MODES 1 and 2:

With less than four reactor coolant pumps in operation, be in HOT STANDBY within 4 hours.

MODES 3, 4** and 5**:

Operation may proceed provided at least one reactor coolant loop is in operation with an associated reactor coolant pump or shutdown cooling pump.# The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.1 The Flow Dependent Selector Switch shall be determined to be in the 4 pump position within 15 minutes prior to making the reactor critical and at least once per 12 hours thereafter.

*See Special Test Exception 3.10.4.

**A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures $\leq$ 275°F unless 1) the pressurizer water volume is less than 600 cubic feet or 2) the secondary water temperature of each steam generator is less than 43°F (31°F when measured by a surface contact instrument) above the coolant temperature in the reactor vessel.

#All reactor coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour, provided no operations are permitted which could cause dilution of the reactor coolant system boron concentration.

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 PSIA ± 1%.

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE shutdown cooling loop into operation.

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 PSIA ± 1%.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.2 Each pressurizer code safety valve shall be demonstrated OPERABLE with a lift setting of 2500 PSIA ± 1%, in accordance with Subsection IWV-3510 of Section XI of the ASME Boiler and Pressure Vessel Code, dated July 1, 1974.

REACTOR COOLANT SYSTEM

RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.3 Two power operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one or more PORV(s) inoperable, within 8 hours either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more block valve(s) inoperable, within 8 hours either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 Each PORV shall be demonstrated OPERABLE:

a. Once per 31 days by performance of a CHANNEL FUNCTIONAL TEST, excluding valve operation, and

b. Once per 18 months by performance of a CHANNEL CALIBRATION.

4.4.3.2 Each block valve shall be demonstrated OPERABLE once per 92 days by operating the valve through one complete cycle of full travel.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a steam bubble and with at least 130 kw of pressurizer heater capacity capable of being supplied by emergency power. The pressurizer level shall be within ± 5% of its programmed value.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

A. With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

B. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4 The pressurizer water level shall be determined to be within ± 5% of its programmed value at least once per 12 hours.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility.

3/4.3.3.4. METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs."

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.3.6 CHLORINE DETECTION SYSTEMS

The operability of the chlorine detection systems ensures that an accidental chlorine release will be detected promptly and the necessary protective actions will be automatically initiated to provide protection for control room personnel. Upon detection of a high concentration of chlorine, the control room emergency ventilation system will automatically isolate the control room and initiate its operation in the recirculation mode of operation to provide the required protection. The chlorine detection systems required by this specification are consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release", February 1975.

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.8 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations".

BASES

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. STARTUP and POWER OPERATION may be initiated and may proceed with one or two reactor coolant pumps not in operation after the setpoints for the Power Level-High, Reactor Coolant Flow-Low, and Thermal Margin/Low Pressure trips have been reduced to their specified values. Reducing these trip setpoints ensures that the DNBR will be maintained above 1.30 during three pump operation and that during two pump operation the core void fraction will be limited to ensure parallel channel flow stability within the core and thereby prevent premature DNB.

A single reactor coolant loop with its steam generator filled above the low level trip setpoint provides sufficient heat removal capability for core cooling while in MODES 2 and 3; however, single failure considerations require plant cooldown if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

The restrictions on starting a Reactor Coolant Pump during MODES 4 and 5 with one or more RCS cold legs $\leq 275^{\circ}F$ are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than $43^{\circ}F$ ($37^{\circ}F$ when measured by a surface contact instrument) above the coolant temperature in the reactor vessel.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 296,000 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the Reactor Coolant System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the pressurizer power operated relief valve or steam dump valves.

3/4.4.3 RELIEF VALVES

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer with the level as programmed ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valve against water relief. The power operated relief valves function to relieve RCS pressure during all design transients. Operation of the power operated relief valve in conjunction with a reactor trip on a Pressurizer-Pressure-High signal, minimizes the undesirable opening of the spring loaded pressurizer code safety valves.

The requirement that 130 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of off-site power condition to maintain natural circulation at HOT STANDBY.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking.

The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.5 GPM, per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 gallon per minute can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Station Superintendent shall be responsible for overall operation of the Millstone Station Site while the Unit Superintendent shall be responsible for operation of the unit. The Station Superintendent and Unit Superintendent shall each delegate in writing the succession to these responsibilities during their absence.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for facility management and technical support shall be as shown on Figure 6.2-1.

FACILITY STAFF

6.2.2 The Facility organization shall be as shown on Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

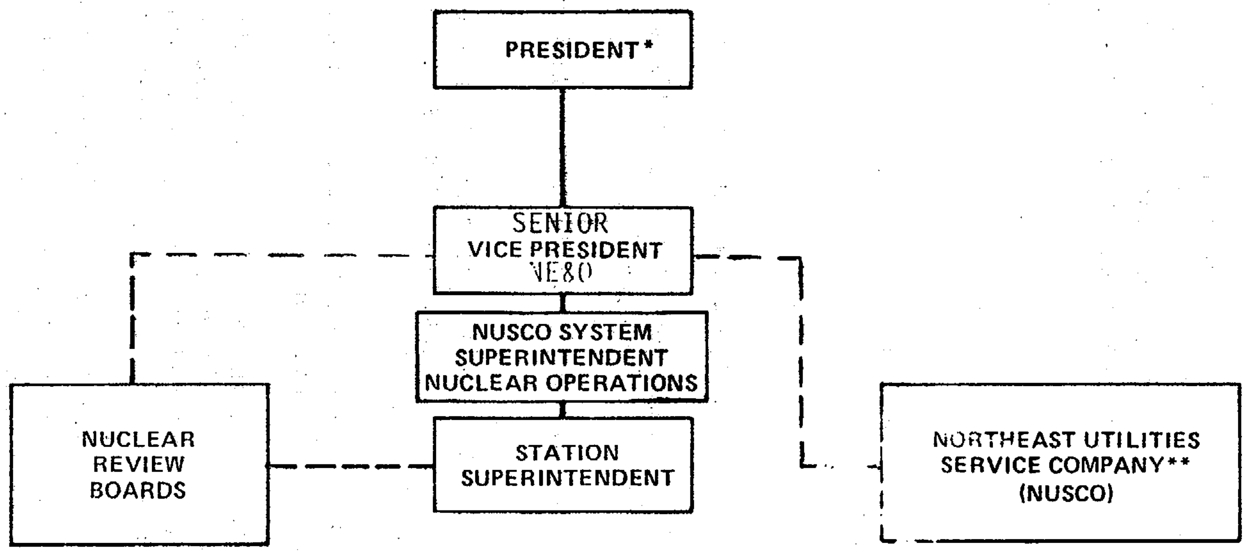
e. All CORE ALTERATIONS after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

f. A site Fire Brigade of at least 3 members shall be maintained onsite at all times. The Fire Brigade shall not include 2 members of the minimum shift crew necessary for safe shutdown of the unit or any personnel required for other essential functions during a fire emergency.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, Revision 1, and (2) the Shift Technical Advisor who shall have a Bachelor's Degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

NORTHEAST NUCLEAR ENERGY COMPANY



* Overall Corporate Responsibility for Fire Protection

** Provides Operating and Engineering Support by Contractual Arrangement

Figure 6.2-1 Offsite Organization for Facility Management and Technical Support

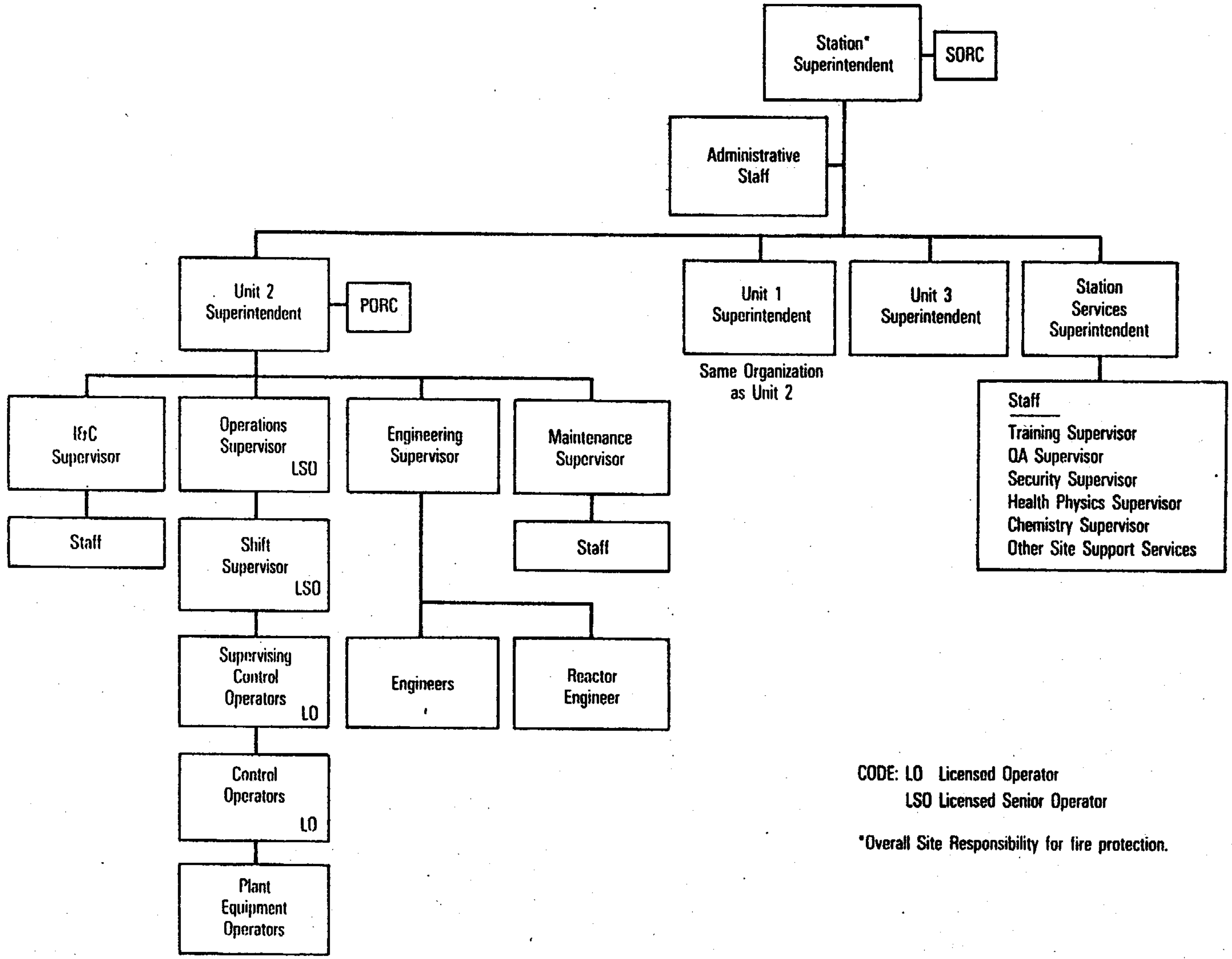


Figure 6.2-2 Facility Organization - Millstone Nuclear Power Station - Unit 2.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

LICENSE CATEGORY	APPLICABLE MODES 1, 2, 3 & 4	APPLICABLE MODES 5 & 6
SOL	1	1*
OL	2	1
Non-Licensed	2	1
Shift Technical Advisor	1	None Required

*Does not include the licensed Senior Reactor or Senior Reactor Operator Limited to Fuel Handling individual supervision CORE ALTERATIONS after the initial fuel loading.

#Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours to accommodate injury or sickness occurring to on duty shift crew members.

g. Records of training and qualification for current members of the plant staff.

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Records of quality assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR Part 50.59.

k. Records of meetings of the PORC, the NRB, the SORC and the SNRB.

l. Records of Environmental Qualification which are covered under the provisions of paragraph 6.13.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

*Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the Radiation Work Permit. The surveillance frequency shall be established by the Health Physics Supervisor.

6.12.2 The requirements of 6.12.1, above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or the Health Physics Supervisor.

6.13 ENVIRONMENTAL QUALIFICATION

6.13.1 By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualifications of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License DPR-65 dated October 24, 1980.

6.13.2 By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.14 SYSTEMS INTEGRITY

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

6.15 IODINE MONITORING

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 66 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 letter dated September 16, 1980, Northeast Nuclear Energy Company (NNECO or the licensee) proposed changes to the Technical Specifications (TS) appended to Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Unit No. 2. The changes involve the incorporation of certain of the TMI-2 Lessons Learned Category "A" requirements. The licensee's request is in direct response to the NRC staff's letter dated July 2, 1980.

BACKGROUND INFORMATION

By our letter dated September 13, 1979, we issued to all operating nuclear power plants requirements established as a result of our review of the TMI-2 accident. Certain of these requirements, designated Lessons Learned Category "A" requirements, were to have been completed by the licensee prior to any operation subsequent to January 1, 1980. Our evaluation of the licensee's compliance with these Category "A" items was attached to our letter to NNECO dated February 25, 1980.

In order to provide reasonable assurance that operating reactor facilities are maintained within the limits determined acceptable following the implementation of the TMI-2 Lessons Learned Category "A" items, we requested that licensees amend their TS to incorporate additional Limiting Conditions for Operation and Surveillance Requirements, as appropriate. This request was transmitted to all licensees on July 2, 1980. Included therein were model specifications that we had determined to be acceptable. Each of the issues identified by the NRC staff and the licensee's response is discussed in the evaluation below.

EVALUATION

2.1.1 Emergency Power Supply Requirements

The pressurizer water level indicators, pressurizer relief and block valves, and pressurizer heaters are important in a post-accident situation. Adequate emergency power supplies add assurance of post-accident functioning of these components. The facility's original design has the requisite emergency power supplies. We find the existing TS already provide appropriate surveillance and actions in the event of component inoperability and are thus acceptable.

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2.1.3.a Direct Indication of PORV and SV Flow

NNECO has provided an acoustic monitoring system downstream of the pressurizer power-operated relief valves (PORVs) and safety valves (SVs) to provide direct indication of flow through any of these valves in the control room. These indications are a diagnostic aid for the operators and provide no automatic action. NNECO has agreed to the TS requirements issued on January 19, 1981 for St. Lucie, Unit No. 1 by Amendment No. 37 to Facility Operating License No. DPR-67. These TS would provide 31-day channel check and 18-month channel calibration requirements. Thus, the TS are acceptable as they meet our July 2, 1980 model TS criteria.

2.1.3.b Instrumentation for Inadequate Core Cooling

The licensee has installed an instrument system to detect the effects of inadequate core cooling. This instrument system, a subcooling meter, receives and processes data from existing plant instrumentation. We previously reviewed this system in our Safety Evaluation dated February 25, 1980. The licensee has agreed to TS with a 31-day channel check and an 18-month channel calibration requirement and appropriate actions to be taken in the event of component inoperability. We conclude the TS are acceptable as they provide adequate surveillance and meet our July 2, 1980 model TS criteria.

2.1.4 Diverse Containment Isolation

NNECO's response indicates that the TS regarding containment isolation valves are adequate in their current form. The existing system has diverse parameters, including high containment pressure and low pressurizer pressure, to be sensed and ensure automatic isolation of nonessential systems under postulated accident conditions. TS Tables 3.3-3 (ESF Instrumentation), 3.3-4 (ESF Actuation Trip Valves), 3.3-5 (ESF Response Times) and 4.3-2 (ESF Surveillance Requirements) provide for the appropriate actions and surveillance requirements. Therefore, we find the current TS acceptable.

2.1.4 Integrity of System Outside Containment

Our request indicated that licensees should propose a license condition to require a periodic System Integrity Measurements Program to prevent the release of significant amounts of radioactivity to the environment via leakage from engineered safety systems and auxiliary systems which are located outside reactor containment. The licensee's present program includes provisions for a preventative maintenance program and periodic visual inspections. The program also includes system leak test measurements at frequencies not to exceed refueling cycle intervals.

In lieu of a license condition, NNECO has agreed to place such a requirement in TS Section 6. Based on our review we find that inclusion of this requirement in the Administrative Controls Section of the TS satisfies our requirement and is acceptable.

2.1.7.a Auto Initiation of Auxiliary Feedwater Systems

This requirement was completed by our issuance of Amendment No. 63, dated January 14, 1981. The related safety evaluation found the TS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) for the auxiliary feedwater system (AFWS) acceptable. Two open items remain to be completed to meet our October 22, 1979 letter giving the Bulletins and Orders Task Force requirements. Our review of the safety grade system that NNECO has installed to automatically initiate AFWS flow will be completed at a later date.

2.1.7.b Auxiliary Feedwater Flow Indications

Our February 25, 1980 evaluation of this item found the control room instrumentation installed at Millstone, Unit No. 2 meets the intent of Item 2.1.7.b.

2.1.8.c Iodine Monitoring

Our request indicated that the licensees should implement a program which will ensure the capability to determine the airborne iodine concentration in areas requiring personnel access under accident conditions. The licensee's present program includes training of personnel, procedures for monitoring and provisions for maintenance of sampling and analysis equipment.

Again, NNECO has agreed to place such a requirement in TS Section 6. Based on our review we find that inclusion of this requirement in the Administrative Controls Section of the TS satisfies our requirement and is acceptable.

2.2.1.b Shift Technical Advisor

Our request indicated that the TS related to minimum shift manning should be revised to reflect the augmentation of Shift Technical Advisor. The licensee's application would add one Shift Technical Advisor to each shift to perform the function of accident assessment during reactor operation. The individual performing this function will have at least a bachelor's degree or equivalent in a scientific or engineering discipline with special training in plant design, and response and analysis of the plant for transients and accidents. Part of the Shift Technical Advisor duties are related to operating experience review function. Based on our review, we find the licensee's submittal satisfies our requirement and is, therefore, 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: April 7, 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. 66 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 as of its date of issuance.

The amendment incorporates certain of the Lessons Learned Category "A" requirements related to the Three Mile Island Accident.

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 application for amendment dated September 16, 1980, (2) Amendment No. 66 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 7th day of April, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

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