December 8, 1978

Dockets Nos. 50-245
and 50-336

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Northeast Nuclear Energy Company
ATTN: Mr. W. G. Counsil, Vice President
Nuclear Engineering and Operations
Post Office Box 270
Hartford, Connecticut 06101

Gentlemen:

The Commission has issued the enclosed Amendments Nos. 56 and 45 to Provisional Operating License No. DPR-21 and Facility Operating License No. DPR-65, respectively, for the Millstone Nuclear Power Station Units Nos. 1 and 2. The amendments consist of changes to the Technical Specifications in accordance with your requests dated July 21, October 4, 12, and 18, November 14, 16, and 21, and December 13 and 15, 1977, and January 12 and 24, February 23, and March 20 and 21, 1978.

The amendments revise the Appendix A Technical Specifications for Unit 2 only by:

- changing the refueling water storage tank sump recirculation actuation setpoint;
- revising the engineered safety features response times;
- modifying the incore detector operability requirements to be more definitive and to remove unnecessary requirements;
- correcting the required number of redundant meteorological monitoring instruments;
- modifying the action requirements for operability of the control room chlorine detectors;
- defining "immediate" in certain ACTION statements and revising other specified time intervals;

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- correcting the pressure at which the safety injection tank isolation valves must be operable;
- increasing the amount of TSP required to neutralize the containment sump after a LOCA;
- changing the wording to indicate that specific doorways in the enclosure building have only one door;
- requiring verification of proper operation of the diesel generator under simulated emergency conditions; and
- adding surveillance requirements for ECCS throttled valves.

The amendments also revise the Appendix A Technical Specifications for both Units 1 and 2 by:

- changing administrative controls to reflect current organizational structure; and
- providing greater flexibility regarding entry into high radiation areas.

Some portions of your proposed Technical Specifications have been modified to meet your own or our requirements. These modifications have been discussed with and agreed to by your staff.

You also submitted an application dated March 14, 1978, to revise Technical Specification Section 4.6.1.2(d) for Unit 2 to clarify the intervals during which Types B and C containment leak rate tests are to be conducted. After discussions with the Office of Inspection and Enforcement and your staff, we have concluded that this change is unnecessary. Consequently, you have withdrawn your application by letter dated July 31, 1978. By letter dated August 21, 1978, you also withdrew a request dated May 31, 1978 for a similar action for Unit 1. We consider these matters closed.

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

Sincerely,

conditioned on incorporation into SE pp. 5 & 6 of noted changes

C-EEB:DOR
GKnighton
12/8/78

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

12-8-78

*SEE PREVIOUS YELLOW FOR CONCURRENCES

Enclosures and cc: See next page

STSG:DOR	C-RS:DOR	ORB#2:DOR	ORB#2:DOR
JMcGough	BGrimes	HSmith*	JShea*
12/8/78	12/8/78	9/25/78	9/22/78

OFFICE	ORB#4:DOR	ORB#4:DOR	PSB:DOR	OELD	C-ORB#4:DOR	C-ORB#2:DOR
SURNAME	RIngram	MConner	GLainas	JGray	RReid	DZiemann
DATE	9/21/78	9/29/78	12/15/78	12/6/78	12/8/78	12/8/78

Enclosures:
1. Amendment No. 56 to DPR-21
2. Amendment No. 45 to DPR-65
3. Safety Evaluation
4. Notice

cc w/enclosures: See next page

cc w/enclosure(s):
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Day, Berry & Howard
Counselors at Law
One Constitution Plaza
Hartford, Connecticut 06103

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

Mr. Albert L. Partridge, First Selectman
Town of Waterford
Hall of Records - 200 Boston Post Road
Waterford, Connecticut 06385

Northeast Nuclear Energy Company
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Branch (AW-459)
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U. S. Environmental Protection Agency
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Region I Office
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Boston, Massachusetts 02203

Waterford Public Library
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Waterford, Connecticut 06385

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Hartford, Connecticut 06101

cc w/encls. & inc.dtd.: 7/21, 10/4,12&18, 11/14,16&21, & 12/13&15/77 & 1/12&24, 2/23, & 3/20&21/78
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

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

DOCKET NO. 50-245

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 1

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 56
License No. DPR-21

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

   A. The applications for amendment by The Connecticut Light and Power Company, The Hartford Electric Light Company, Western Massachusetts Electric Company, and Northeast Nuclear Energy Company (the licensees), dated March 20 and 21, 1978, 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 applications, the provisions of the Act, and the rules and regulations of the Commission;

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

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Provisional Operating License No. DPR-21 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 56, are hereby incorporated in the license. The licensees 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

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 8, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 56

PROVISIONAL OPERATING LICENSE NO. DPR-21

DOCKET NO. 50-245

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. Overleaf pages are provided for document completeness.

REMOVE	INSERT
6-1	6-1
6-2	6-2
6-3	6-3
6-5	6-5
6-6	6-6
6-7	6-7
6-8	6-8
6-9	6-9
6-10	6-10 *
6-12	6-12
6-15	6-15
6-16	6-16
6-17	6-17
6-23	6-23
--	6-24
6-25	--

*Section 6.5.3.3 (Alternates) has been deleted because the revised Section 6.5.3.2 (Composition) encompasses the alternate members.

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 in 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.

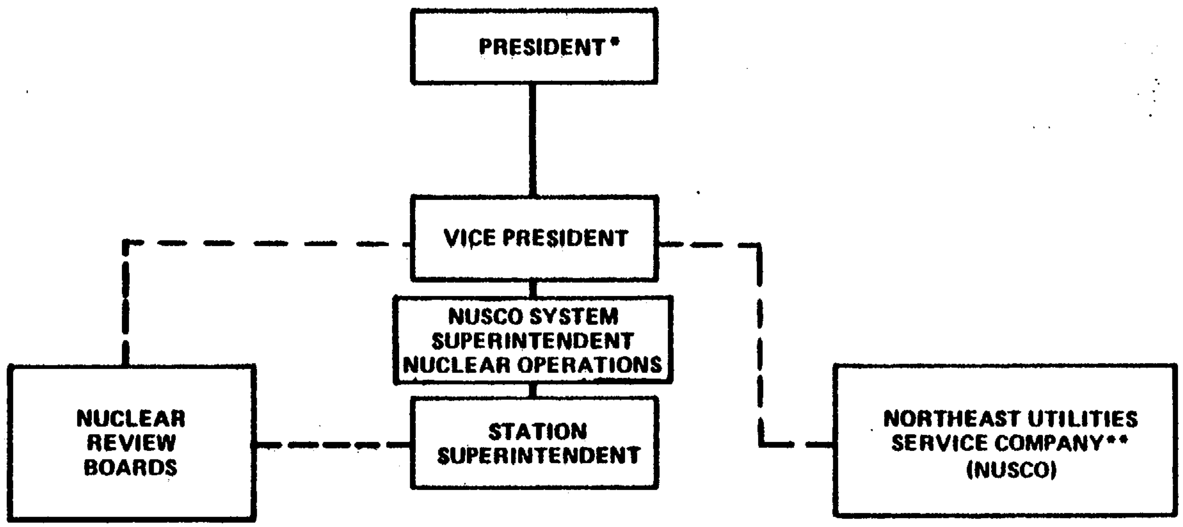
f. A Fire Brigade of 3 members shall be maintained onsite at all times. The Fire Brigade shall not include the minimum shift crew necessary for safe shutdown of the Unit (2 members) or any personnel required for other essential functions during a fire emergency.*

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 the Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, Revision 1, after January 1, 1978.

*To be effective by March 1, 1978

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

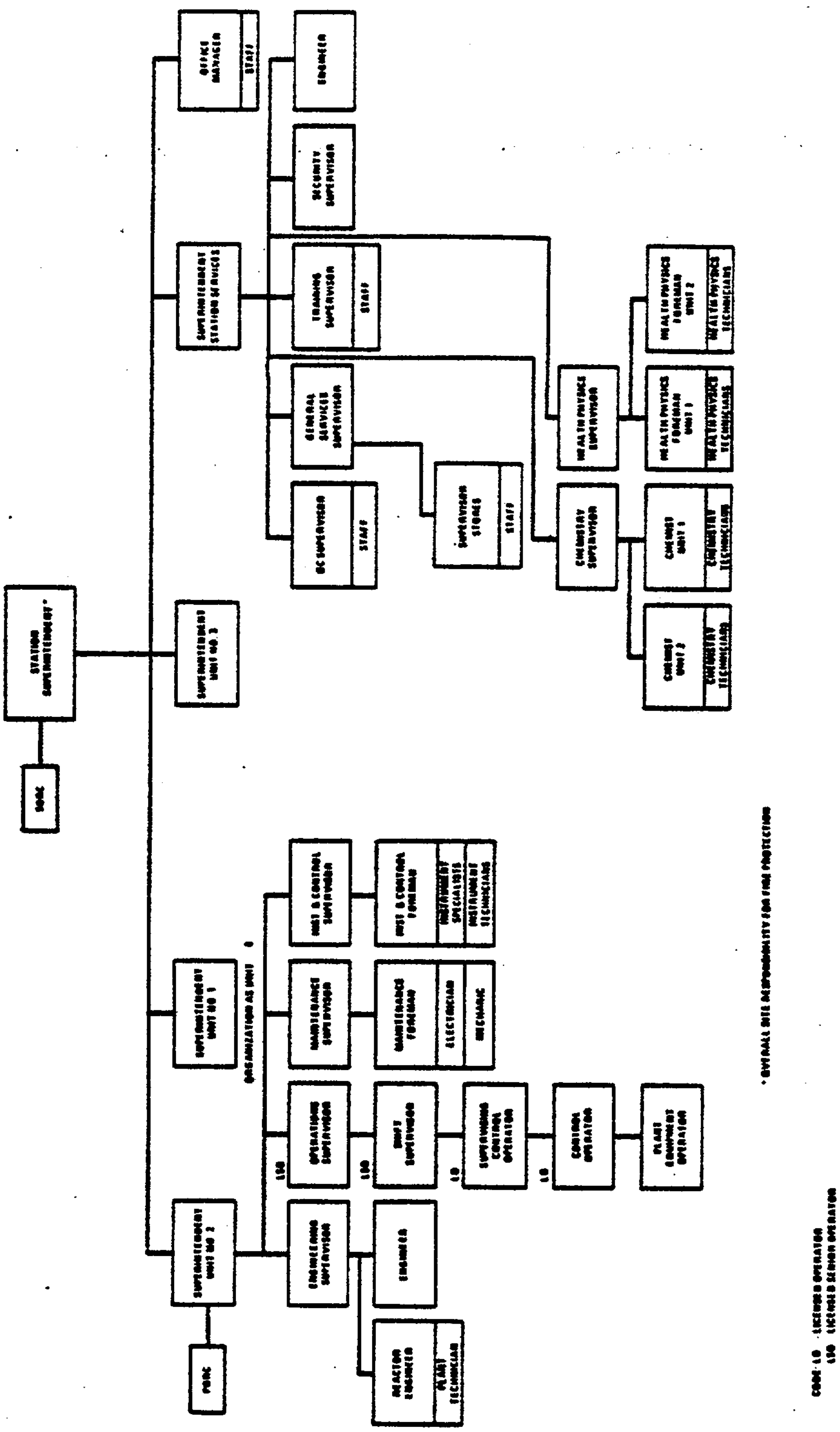


Figure 6.2.2 Facility Organization – Millstone Nuclear Power Station

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

LICENSE CATEGORY	APPLICABLE MODES 1, 2, & 3**	APPLICABLE MODES 4 & 5##
SOL	1	1*
OL	2	1
Non-Licensed	2	1

*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.

**Reactor Mode Switch Position is in RUN (any average coolant temperature) STARTUP/HOT STANDBY (any average coolant temperature), and HOT SHUTDOWN (average coolant temperature greater than 212°F), respectively.

##Reactor Mode Switch Position is in SHUTDOWN (average coolant temperature less than 212°F) and REFUELING (average coolant temperature less than 212°F), respectively.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Station Superintendent and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Training Supervisor and shall meet or exceed the requirements of Section 27 of the NFPA Code - 1976, except that Fire Brigade training sessions shall be held at least quarterly.#

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC)

FUNCTION

6.5.1.1 The PORC shall function to advise the Unit Superintendent on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PORC shall be composed of the:

Chairman:	Unit Superintendent
Vice Chairman & Member:	Operations Supervisor
Member:	Maintenance Supervisor
Member:	Instrument and Control Supervisor
Member:	Reactor Engineer
Member:	Senior Engineer or Startup Supervisor*
Member:	Chemistry Supervisor or Station Services Superintendent of Service Group Supervisor or Health Physics Supervisor
Member:	Staff Engineer**

ALTERNATES

6.5.1.3 Alternate members shall be appointed in writing by the PORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate in PORC activities at any one time.

*When position is staffed.

**The Staff Engineer member of the PORC shall have an academic degree in engineering or physical science field; and in addition, shall have a minimum of five years technical experience, of which a minimum of three years shall be in the nuclear power plant industry.

#To be effective by March 1, 1978.

MEETING FREQUENCY

6.5.1.4 The PORC shall meet at least once per calendar month and as convened by the PORC Chairman.

QUORUM

6.5.1.5 A quorum of the PORC shall consist of the Chairman or Vice Chairman or Station Superintendent and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The PORC shall be responsible for:

a. Review of 1) all procedures, except common site procedures, required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Unit Superintendent to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to Sections 1.0 - 5.0 of these Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications and preparation and forwarding of a report covering evaluation and recommendations to prevent recurrence to the System Superintendent Nuclear Operations and to the Chairman of the Nuclear Review Board.

f. Review of events requiring 24 hour notification to the Commission.

g. Review of facility operations to detect potential safety hazards.

h. Performance of special reviews and investigations and reports thereon as requested by the Chairman of the Nuclear Review Board.

i. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a) through (e) above consitutes an unreviewed safety question.

AUTHORITY

6.5.1.7 The PORC shall:

a. Recommend to the Unit Superintendent written approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

b. Provide immediate written notification to the Station Superintendent, System Superintendent Nuclear Operations and the Nuclear Review Board of disagreement between the PORC and the Unit Superintendent; however, the Unit Superintendent shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The PORC shall maintain written minutes of each meeting and copies shall be provided to the Station Superintendent, System Superintendent Nuclear Operations and Chairman of the Nuclear Review Board.

6.5.2 SITE OPERATIONS REVIEW COMMITTEE (SORC)

FUNCTION

6.5.2.1 The SORC shall function to advise the Station Superintendent on all matters related to nuclear safety of the entire Millstone Station Site.

COMPOSITION

6.5.2.2 The SORC shall be composed of the:

Chairman:	Station Superintendent
Member:	Unit 1 Superintendent
Member:	Unit 2 Superintendent
Member:	Unit 3 Superintendent
Member:	Station Services Superintendent
Member:	Designated Member of Unit 1 PORC
Member:	Designated Member of Unit 2 PORC
Member:	Designated Member of Unit 3 PORC

ALTERNATES

6.5.2.3 Alternate members shall be appointed in writing by the SORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate in SORC activities at one time.

MEETING FREQUENCY

6.5.2.4 The SORC shall meet at least once per six months and as convened by the SORC Chairman.

QUORUM

6.5.2.5 A quorum of the SORC shall consist of the Chairman and four members including alternates.

RESPONSIBILITIES

6.5.2.6 The SORC shall be responsible for:

a. Review of 1) all common site procedures required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Station Superintendent to affect site nuclear safety.

b. Review of all proposed changes to Section 6.0 "Administrative Controls" of these Technical Specifications.

c. Performance of special reviews and investigations and reports as requested by the Chairman of the Site Nuclear Review Board.

d. Review of the Plant Security Plan and implementing procedures and shall submit changes to the Chairman of the Site Nuclear Review Board.

e. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the Site Nuclear Review Board.

f. Review of all common site proposed tests and experiments that affect nuclear safety.

g. Review of all common site proposed changes or modifications to systems or equipment that affect nuclear safety.

RESPONSIBILITIES (Continued)

h. Render determinations in writing or meeting minutes with regard to whether or not each item considered under 6.5.2.6(a) through (g) above constitutes an unreviewed safety question.

AUTHORITY

6.5.2.7 The SORC shall:

a. Recommend to the Station Superintendent written approval or disapproval in meeting minutes of items considered under 6.5.2.6(a) through (g) above.

b. Provide immediate written notification or meeting minutes to the Superintendent of Nuclear Production and the Site Nuclear Review Board of disagreement between the SORC and the Station Superintendent; however, the Station Superintendent shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.2.8 The SORC shall maintain written minutes of each meeting and copies shall be provided to the System Superintendent Nuclear Operations and Chairman of the Site Nuclear Review Board.

6.5.3 NUCLEAR REVIEW BOARD (NRB)

FUNCTION

6.5.3.1 The NRB shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

COMPOSITION

6.5.3.2 The Chairman, members and alternate members of the NRB shall be appointed in writing by the Vice President, Engineering and Operations, and shall have an academic degree in an engineering or physical science field; and, in addition, shall have a minimum of five years technical experience, of which a minimum of three years shall be in their respective fields of expertise. No more than two alternates shall participate as voting members in NRB activities at any one time.

ALTERNATES

6.5.3.3 (Deleted)

CONSULTANTS

6.5.3.4 Consultants shall be utilized as determined by the NRB Chairman to provide expert advice to the NRB.

MEETING FREQUENCY

6.5.3.5 The NRB shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.3.6 The minimum quorum of the NRB necessary for the performance of the NRB review and audit functions of these technical specifications shall consist of the Chairman or his designated alternate and at least four members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW

6.5.3.7 The NRB shall review:

a. The safety evaluation for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes in Sections 1.0 - 5.0 of these Technical Specifications or licenses.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. REPORTABLE OCCURRENCE, as defined in Revision 4 to Regulatory Guide 1.16.

h. Indications of a significant unanticipated deficiency, affecting nuclear safety, in some aspect of design or operation of safety related structures, systems or components.

i. Reports and meetings minutes of the PORC.

AUDITS

6.5.3.8 Audits of facility activities shall be performed under the cognizance of the NRB. These audits shall encompass:

a. The conformance of facility operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per year.

AUDITS (Continued)

b. The performance, training and qualifications of the entire facility staff at least once per year.

c. The results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per six months.

d. Any other area of facility operation considered appropriate by the NRB or the Vice President System Production.

AUTHORITY

6.5.3.9 The NRB shall report to and advise the Vice President Nuclear Engineering and Operations on those areas of responsibility specified in Sections 6.5.3.7 and 6.5.3.8.

RECORDS

6.5.3.10 Records of NRB activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each NRB meeting shall be prepared, approved and forwarded to the Vice President Nuclear Engineering and Operations within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.3.7 above, shall be prepared, approved and forwarded to the Vice President Nuclear Engineering and Operations within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.3.8 above, shall be forwarded to the Vice President Nuclear Engineering and Operations and to the management positions responsible for the areas audited within 30 days after completion of the audit.

AUTHORITY

6.5.4.9 The SNRB report to and advise the Vice President Nuclear Engineering and Operations on those areas of responsibility specified in Sections 6.5.4.7 and 6.5.4.8.

RECORDS

6.5.4.10 Records of SNRB activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each SNRB meeting shall be prepared, approved and forwarded to the Vice President Nuclear Engineering and Operations within 14 days following each meeting.

b. Reports of reviews encompassed by, Section 6.5.4.7 above, shall be prepared, approved and forwarded to the Vice President Nuclear Engineering and Operations within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.4.8 above, shall be forwarded to the Vice President Nuclear Engineering and Operations and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE Report requiring 24 hour notification to the Commission shall be reviewed by the PORC and submitted to the NRB and the System Superintendent Nuclear Operations.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The provisions of 10 CFR 50.36(c)(1)(i) shall be complied with immediately.

SAFETY LIMIT VIOLATION (Continued)

b. The Safety Limit violation shall be reported to the Commission, the System Superintendent Nuclear Operations and to the NRB immediately.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the NRB and the System Superintendent Nuclear Operations within 10 days of the violations.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972.

b. Refueling operations.

c. Surveillance activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed by the PORC/SORC, as applicable, and approved by the Unit Superintendent/ Station Superintendent prior to implementation and reviewed periodically as set forth in each document.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the PORC/SORC, as applicable, and approved by the Unit Superintendent/Station Superintendent within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

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.

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.

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

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

DOCKET NO. 50-336

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 45
License No. DPR-65

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

   A. The applications for amendment by The Connecticut Light and Power Company, The Hartford Electric Light Company, Western Massachusetts Electric Company, and Northeast Nuclear Energy Company (the licensees), dated July 21, October 4, 12, and 18, November 14, 16, and 21, and December 13 and 15, 1977, and January 12 and 24, February 23, and March 20 and 21, 1978, 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 applications, the provisions of the Act, and the rules and regulations of the Commission;

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

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 45, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 8, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 45

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

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

Pages	Pages
XVI	6-1
B 2-6	6-2
3/4 1-24	6-3
3/4 1-24a (deleted)	6-5
3/4 3-19	6-6
3/4 3-21	6-7
3/4 3-22	6-8
3/4 3-30	6-9
3/4 3-31	6-10
3/4 3-37	6-11
3/4 3-38	6-12
3/4 3-42	6-15
3/4 4-3	6-16
3/4 4-17	6-17
3/4 4-21	6-23
3/4 5-2	6-24 (added)
3/4 5-5	
3/4 5-6	
3/4 5-6a (added)	
3/4 5-7	
3/4 6-28	
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INDEX

ADMINISTRATIVE CONTROLS

ADMINISTRATIVE CONTROLS

Reactor Coolant Flow-Low (Continued)

operation of the reactor at reduced power if one or two reactor coolant pumps are taken out of service. The low-flow trip setpoints and Allowable Values for the various reactor coolant pump combinations have been derived in consideration of instrument errors and response times of equipment involved to maintain the DNBR above 1.30 under normal operation and expected transients. For reactor operation with only two or three reactor coolant pumps operating, the Reactor Coolant Flow-Low trip setpoints, the Power Level-High trip setpoints, and the Thermal Margin/Low Pressure trip setpoints are automatically changed when the pump condition selector switch is manually set to the desired two- or three-pump position. Changing these trip setpoints during two and three pump operation prevents the minimum value of DNBR from going below 1.30 during normal operational transients and anticipated transients when only two or three reactor coolant pumps are operating.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated concurrently with a safety injection. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setting of 500 psia is sufficiently below the full-load operating point of 815 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used with an uncertainty factor of ± 22 psi in the accident analyses.

Steam Generator Water Level - Low

The Steam Generator Water Level-Low Trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the design pressure of the reactor coolant system will not be exceeded. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to provide a margin of more than 10 minutes before auxiliary feedwater is required.

Local Power Density-High

The Local Power Density-High trip, functioning from AXIAL SHAPE INDEX monitoring, is provided to ensure that the peak local power density in the fuel which corresponds to fuel centerline melting will not occur as a consequence of axial power maldistributions. A reactor trip is initiated whenever the AXIAL SHAPE INDEX exceeds the allowable limits of Figure 2.2-2. The AXIAL SHAPE INDEX is calculated from the upper and lower ex-core neutron detector channels. The calculated setpoints are generated as a function of THERMAL POWER level with the allowed CEA group position being inferred from the THERMAL POWER level. The trip is automatically bypassed below 15 percent power.

The maximum AZIMUTHAL POWER TILT and maximum CEA misalignment permitted for continuous operation are assumed in generation of the setpoints. In addition, CEA group sequencing in accordance with the Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

Thermal Margin/Low Pressure

The Thermal Margin/low Pressure trip is provided to prevent operation when the DNBR is less than 1.30.

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POSITION INDICATOR CHANNELS

LIMITING CONDITION FOR OPERATION

3.1.3.3 All shutdown and regulating CEA reed switch position indicator channels and CEA pulse counting position indicator channels shall be OPERABLE and capable of determining the absolute CEA positions within ± 3 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. Deleted.

b. With a maximum of one reed switch position indicator channel per group or one (except as permitted by ACTION item d. below) pulse counting position indicator channel per group inoperable and the CEA(s) with the inoperable position indicator channel partially inserted, within 4 hours either:

   1. Restore the inoperable position indicator channel to OPERABLE status, or

   2. Be in HOT STANDBY, or

   3. Reduce THERMAL POWER to ≤ 70% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination; if negative reactivity insertion is required to reduce THERMAL POWER, boration shall be used. Operation at or below this reduced THERMAL POWER level may continue provided that within the next 4 hours either:

      a) The CEA group(s) with the inoperable position indicator is fully withdrawn while maintaining the withdrawal sequence required by Specification 3.1.3.6 and when this CEA group reaches its fully withdrawn position, the "Full Out" limit of the CEA with the inoperable position indicator is actuated and verifies this CEA to be fully withdrawn. Subsequent to fully withdrawing this CEA group(s), the THERMAL POWER level may be returned to a level consistent with all other applicable specifications; or

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	$\leq$ 5 psig	$\leq$ 5 psig
d. Pressurizer Pressure - Low	$\geq$ 1600 psia	$\geq$ 1600 psia
6. CONTAINMENT SUMP RECIRCULATION (SRAS)		
a. Manual SRAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Storage Tank - Low	48 $\pm$ 18 inches above tank bottom	48 $\pm$ 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	9100 cpm	9100 cpm
Particulate Activity	1.0×10^6 cpm/hr	1.0×10^6 cpm/hr

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 $\pm$ 2.0 second time delay	$\geq$ 3700 volts with an 8.0 $\pm$ 2.0 second time delay

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
2. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	
1) High Pressure Safety Injection	$\le$ 30.0*/5.0**
2) Low Pressure Safety Injection	$\le$ 50.0*/5.0**
b. Containment Isolation	$\le$ 7.5
c. Enclosure Building Filtration System	$\le$ 50.0*/35.0**

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	$\leq$ 30.0*/5.0**
2) Low Pressure Safety Injection	$\leq$ 50.0*/5.0**
b. Containment Isolation	$\leq$ 7.5
c. Enclosure Building Filtration System	$\leq$ 50.0*/35.0**
4. Containment Pressure--High-High	
a. Containment Spray	$\leq$ 35.6*(1)/35.6**(1)
5. Containment Radiation-High	
a. Containment Purge Valves Isolation	$\leq$ Counting period plus 7.5
6. Steam Generator Pressure-Low	
a. Main Steam Isolation	$\leq$ 6.9
b. Feedwater Isolation	$\leq$ 60
7. Refueling Water Storage Tank-Low	
a. Containment Sump Recirculation	$\leq$ 120

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.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Spent Fuel Storage				
Criticality Monitor	S	R	M	*
Ventilation System Isolation	S	R	M	#
2. PROCESS MONITORS				
a. Containment Atmosphere-Particulate	S	R	M	ALL MODES
b. Containment Atmosphere-Gaseous	S	R	M	ALL MODES
c. Spent Fuel Storage-Particulate	S	R	M	*
d. Spent Fuel Storage-Gaseous	S	R	M	*

*With fuel in storage building;

#With irradiated fuel in the storage pool.

X

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with at least one OPERABLE detector segment in each core quadrant on each of the four axial elevations containing incore detectors and as further specified below:

a. For monitoring the AZIMUTHAL POWER TILT:

At least two quadrant symmetric incore detector segment groups at each of the four axial elevations containing incore detectors in the outer 184 fuel assemblies with sufficient OPERABLE detector segments in these detector groups to compute at least two AZIMUTHAL POWER TILT values at each of the four axial elevations containing incore detectors.

b. For recalibration of the excore neutron flux detection system:

1. At least 75% of all incore detector segments,

2. A minimum of 9 OPERABLE incore detector segments at each detector segment level, and

3. A minimum of 2 OPERABLE detector segments in the inner 109 fuel assemblies and 2 OPERABLE segments in the outer 108 fuel assemblies at each segment level.

c. For monitoring the UNRODDED PLANAR RADIAL PEAKING FACTOR, the UNRODDED INTEGRATED RADIAL PEAKING FACTOR, or the linear heat rate:

1. At least 75% of all incore detector locations,

2. A minimum of 9 OPERABLE incore detector segments at each detector segment level, and

3. A minimum of 2 OPERABLE detector segments in the inner 109 fuel assemblies and 2 OPERABLE segments in the outer 108 fuel assemblies at each segment level.

An OPERABLE incore detector segment shall consist of an OPERABLE rhodium detector constituting one of the segments in a fixed detector string.

An OPERABLE incore detection location shall consist of a string in which at least three of the four incore detector segments are OPERABLE.

LIMITING CONDITION FOR OPERATION (Continued)

An OPERABLE quadrant symmetric incore detector segment group shall consist of a minimum of three OPERABLE rhodium incore detector segments in 90° symmetric fuel assemblies.

APPLICABILITY: When the incore detection system is used for:

a. Monitoring the AZIMUTHAL POWER TILT,

b. Recalibration of the excore neutron flux detection system, or

c. Monitoring the UNRODDED PLANAR RADIAL PEAKING FACTOR, the UNRODDED INTEGRATED RADIAL PEAKING FACTOR, or the linear heat rate.

ACTION:

With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use and at least once per 7 days thereafter when required for:

1. Monitoring the AZIMUTHAL POWER TILT.

2. Recalibration of the excore neutron flux detection system.

3. Monitoring the UNRODDED PLANAR RADIAL PEAKING FACTOR, the UNRODDED INTEGRATED RADIAL PEAKING FACTOR, or the linear heat rate.

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The neutron detectors shall be calibrated prior to installation in the reactor core.

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation channels shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: ALL MODES.

ACTION:

a. With the number of OPERABLE seismic monitoring channels less than required by Table 3.3-7, restore the inoperable channel(s) to OPERABLE status within 30 days. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With one or more seismic monitoring channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the system to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.3.3.3 Each of the above seismic monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-4.

Table 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	LOCATION	INSTRUMENT MINIMUM ACCURACY	MINIMUM CHANNELS OPERABLE
1. WIND SPEED			
a.	Nominal Elev. 142 ft.	± 0.22 m/sec*	1
b.	Nominal Elev. 374 ft.	± 0.22 m/sec*	1
2. WIND DIRECTION			
a.	Nominal Elev. 142 ft.	± 5°	1
b.	Nominal Elev. 374 ft.	± 5°	1
3. AIR TEMPERATURE - DELTA T			
a.	Nominal Elev. 142 ft.	± 0.18°F	1
b.	Nominal Elev. 347 ft.	± 0.18°F	1

* Starting speed of anemometer shall be < 0.45 m/sec.

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. WIND SPEED		
a. Nominal Elev. 142	D	SA
b. Nominal Elev. 374	D	SA
2. WIND DIRECTION		
a. Nominal Elev. 142	D	SA
b. Nominal Elev. 374	D	SA
3. AIR TEMPERATURE - DELTA T		
a. Nominal Elev. 142	D	SA
b. Nominal Elev. 374	D	SA

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wide Range Logarithmic Neutron Flux	M	N.A.
2. Reactor Trip Breaker Indication	M	N.A.
3. Reactor Cold Leg Temperature	M	R
4. Pressurizer Pressure		
a. Low Range	M	R
b. High Range	M	R
5. Pressurizer Level	M	R
6. Steam Generator Level	M	R
7. Steam Generator Pressure	M	R

INSTRUMENTATION

CHLORINE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.3.3.6 Two separate and independent chlorine detection systems, with their alarm/trip setpoints adjusted to actuate at a chlorine concentration of $\leq$ 5 ppm, shall be OPERABLE with a detector located in each control room outside air intake duct.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one chlorine detection system inoperable, restore the inoperable detection system to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

b. With no chlorine detection system OPERABLE, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each chlorine detection system shall be verified energized at least once per 12 hours and demonstrated OPERABLE by performance of a CHANNEL CALIBRATION at least once per 18 months.

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 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

4.4.3 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.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a steam bubble.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the pressurizer inoperable, be in HOT SHUTDOWN within 8 hours.

SURVEILLANCE REQUIREMENTS

4.4.4 Not applicable.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4-2 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 100°F in any one hour period with T_{avg} above 300°F and a maximum cooldown of 20°F in any one hour period with T_{avg} below 300°F.

c. A maximum temperature change of 5°F in any one hour period, during hydrostatic testing operations above system design pressure.

APPLICABILITY: MODES 1, 2*, 3, 4 and 5.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psia, respectively, within the following 30 hours.

*See Special Test Exception 3.10.3.

SURVEILLANCE REQUIREMENTS

4.4.9.1

a. The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per hour during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

b. The Reactor Coolant System temperature and pressure conditions shall be determined to be to the right of the criticality limit line within 15 minutes prior to making the reactor critical.

c. The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals shown in Table 4.4-3. The results of these examinations shall be used to update Figure 3.4-2.

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 200°F in any one hour period, and

c. A maximum spray water temperature differential of 350°F.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperature and spray water temperature differential shall be determined to be within the limits at least once per hour during system heatup or cooldown.

STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of components (except steam generator tubes) identified in Section 1.2.14 of the FSAR as Safety Class 1 components and of the steam generator secondary side circumferential shell welds shall be maintained at a level consistent with the acceptance criteria in Specification 4.4.10.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the structural integrity of any of the above components not conforming to the above requirements and $T_{avg} > 200°F$, either immediately isolate the affected component or be in COLD SHUTDOWN within the next 36 hours.

b. With the structural integrity of any of the above components not conforming to the above requirements and the unit in COLD SHUTDOWN, restore the structural integrity of the affected component to within its limits prior to increasing the Reactor Coolant System temperature above the minimum temperature required by NDT considerations.

SURVEILLANCE REQUIREMENTS

4.4.10 The following inspection program shall be performed:

a. Inservice Inspections The structural integrity of the Safety Class 1 components shall be demonstrated by verifying their acceptability per the requirements of Articles IS-200 and IS-500 of Section XI of the ASME Boiler and Pressure Vessel Code, dated July 1971, including the Summer 1971 Addendum, as outlined by the inspection program shown in Table 4.4-4.

The structural integrity of the steam generator secondary side circumferential shell welds shall be demonstrated by verifying their acceptability per the requirements of Tables ISC-261, ISC-251 and Section ISC-240 of Section XI of the ASME Boiler and Pressure Vessel Code, Winter 1972 Addendum, as outlined by the inspection program shown in Table 4.4-4.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system safety injection tank shall be OPERABLE with:

a. The isolation valve open and the power to the valve operator removed,

b. Between 1107 and 1170 cubic feet of borated water (equivalent to between 55% and 58% of total tank volume, respectively),

c. A minimum boron concentration of 1720 PPM, and

d. A nitrogen cover-pressure of between 200 and 250 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within one hour or be in HOT SHUTDOWN within the next 8 hours.

b. With one safety injection tank inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the water level and nitrogen cover-pressure in the tanks, and

2. Verifying that each safety injection tank isolation valve is open.

* With pressurizer pressure $\geq$ 1750 psia.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and at each solution volume increase of $\geq$ 1% of tank volume by verifying the boron concentration of the safety injection tank solution.

c. At least once per 31 days by verifying that the closing coil in the valve breaker cubicle is removed.

d. Verifying at least once per 18 months that the safety injection tank isolation valves open automatically before the Reactor Coolant System pressure exceeds 1750 psia and on a safety injection signal.

SURVEILLANCE REQUIREMENTS (Continued)

8. Verifying that the following valves are in the indicated position with power to the valve operator removed:

Valve Number	Valve Function	Valve Position
2-SI-306	Shutdown Cooling Flow Control	Open
2-SI-659	SRAS Recirc.	Open*
2-SI-660	SRAS Recirc.	Open*
2-CH-434	Thermal Bypass	Closed**

b. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. Of the areas affected within containment at the completion of containment entry when CONTAINMENT INTEGRITY is established.

c. At least once per 18 months by:

1. Verifying automatic isolation of the shutdown cooling system from the Reactor Coolant System when the Reactor Coolant System pressure is above 300 psia.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

3. Verifying that a minimum total of 110 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

4. Verifying that when a representative sample of 0.35 + 0.05 lbs of TSP from a TSP storage basket is submerged, without agitation, in 50 $\pm$ 5 gallons of 180 $\pm$ 10°F borated water from the RWST, the pH of the mixed solution is raised to $\geq$ 6 within 4 hours.

*To be closed prior to recirculation following LOCA.
**2-CH-434, a manual valve, shall be locked closed.

d. At least once per 18 months, during shutdown, by cycling each power operated valve in the subsystem flow path not testable during plant operation through one complete cycle of full travel.

e. By a visual verification that each of the throttle valves in Table 4.5-1 will open to the correct position. This verification shall be performed:

   1. Within 4 hours following the completion of each valve stroking operation,

   2. Immediately prior to returning the valve to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator or its control circuit, or

   3. At least once per 18 months.

f. By conducting a flow balance verification immediately prior to returning to service any portion of a subsystem after the completion of a modification that could alter system flow characteristics. The injection leg flow rate shall be as follows:

   1. HPSI Headers - the sum of the three lowest injection flows must be $\geq$ 471 gpm. The sum of the four injection flows must be $\leq$ 675 gpm.

   2. LPSI Header - the sum of the three lowest injection flows must be $\geq$ 2370 gpm. The sum of the four injection flows must be $\leq 4500 + \left[\frac{\text{RWST level (\%)} - 10(\%)}{90\%} \times 200 \right]$

TABLE 4.5-1

ECCS THROTTLED VALVES

1.	2-SI-617	"A" HPSI Header - Loop 1A Injection
2.	2-SI-627	"A" HPSI Header - Loop 1B Injection
3.	2-SI-637	"A" HPSI Header - Loop 2A Injection
4.	2-SI-647	"A" HPSI Header - Loop 2B Injection
5.	2-SI-616	"B" HPSI Header - Loop 1A Injection
6.	2-SI-626	"B" HPSI Header - Loop 1B Injection
7.	2-SI-636	"B" HPSI Header - Loop 2A Injection
8.	2-SI-646	"B" HPSI Header - Loop 2B Injection
9.	2-SI-615	LPSI Header - Loop 1A Injection
10.	2-SI-625	LPSI Header - Loop 1B Injection
11.	2-SI-635	LPSI Header - Loop 2A Injection
12.	2-SI-645	LPSI Header - Loop 2B Injection

ECCS SUBSYSTEMS - $T_{avg} < 300°F$

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE high-pressure safety injection pump, and

b. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a sump recirculation actuation signal.

APPLICABILITY: MODES 3* and 4.

ACTION:

a. With no ECCS subsystem OPERABLE, restore at least one ECCS subsystem to OPERABLE status within one hour or be in COLD SHUTDOWN within the next 20 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.3 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

*With pressurizer pressure < 1750 psia.

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank shall be OPERABLE with:

a. A minimum contained volume of 370,000 gallons of borated water,

b. A minimum boron concentration of 1720 ppm,

c. A minimum water temperature of 50°F when in MODES 1 and 2, and

d. A minimum water temperature of 35°F when in MODES 3 and 4.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore tank to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the water level in the tank, and

2. Verifying the boron concentration of the water.

b. When in MODES 3 and 4, at least once per 24 hours by verifying the RWST temperature is $\geq$ 35°F when the RWST ambient air temperature is < 35°F.

c. When in MODES 1 and 2, at least once per 24 hours by verifying the RWST temperature is $\geq$ 50°F when the RWST ambient air temperature is < 50°F.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is $\leq$ 6 inches Water Gauge while operating the ventilation system at a flow rate of 9000 cfm $\pm$ 10%.

2. Verifying that the air flow distribution to each HEPA filter and charcoal adsorber is within $\pm$ 20% of the averaged flow per unit.

3. Verifying that the filtration system starts automatically on a Enclosure Building Filtration Actuation Signal (EBFAS).

4. Verifying that each system produces a negative pressure of $\geq$ 0.25 inches W. G. in the Enclosure Building Filtration Region within 1 minute after an Enclosure Building Filtration Actuation Signal (EBFAS).

CONTAINMENT SYSTEMS

ENCLOSURE BUILDING INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.2 ENCLOSURE BUILDING INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the switchyard, and

b. Two separate and independent diesel generators each with a separate fuel oil supply tank containing a minimum of 12,000 gallons of fuel.

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

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1 and 4.8.1.1.2.a.2 within one hour and at least once per 8 hours thereafter; restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 36 hours.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1 and 4.8.1.1.2.a.2 within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in COLD SHUTDOWN within the next 36 hours. Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in COLD SHUTDOWN within the next 36 hours.

ACTION (Continued)

c. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.2 within one hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 4 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in COLD SHUTDOWN within the next 36 hours.

d. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1 within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in COLD SHUTDOWN within the next 36 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Two physically independent circuits between the offsite transmission network and the switchyard shall be determined OPERABLE at least once per 24 hours by verifying correct breaker alignments and indicated power availablity.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the fuel oil supply tank,

2. Verifying the diesel starts from ambient condition and accelerates to $\geq$ 90% of rated speed and to $\geq$ 97% of rated voltage in $\leq$ 20 seconds.

3. Verifying the generator is synchronized, loaded to $\geq$ 1300 kw in $\leq$ 60 seconds, and operates for $\geq$ 60 minutes.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank obtained in accordance with ASTM-D270-65, is within the acceptable limits specified in Table 1 of ASTM D975-74 when checked for viscosity, water and sediment.

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

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2. Verifying that the automatic sequence time delay relays are OPERABLE with the following settings:

Sequence Step	Time After Closing of Diesel Generator Output Breaker (Seconds)
1	1.5 - 2.2 seconds
2	7.0 - 8.4 seconds
3	12.5 - 14.6 seconds
4	18.0 - 20.8 seconds

3. Verifying the generator capability to reject a load of $\geq$ 250 kw and maintain voltage at 4160 $\pm$ 500 volts and frequency at 60 $\pm$ 3 Hz.

4. Verifying the generator capability to reject a load of 1300 Kw without exceeding the overspeed trip setpoint (15% above nominal).

5. Simulating a loss of offsite power in conjunction with a safety injection actuation signal, and:

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses,

b) Verifying the diesel starts from ambient condition on the autostart signal, energizes the emergency busses with permanently connected loads, energizes the auto-connected emergency loads through the load sequencer and operates for $\geq$ 5 minutes while its generator is loaded with the emergency loads.

SURVEILLANCE REQUIREMENTS (Continued)

c) Verifying that on the safety injection actuation signal, all diesel generator trips, except engine overspeed, generator differential current, voltage restraint overcurrent, and low lube oil pressure (2 out of 3) are automatically bypassed.

d) Verifying that on a simulated loss of the diesel generator, the loads are shed from the emergency busses, the emergency busses are reenergized with permanently connected through the load sequencer, and the diesel operates for $\geq$ 5 minutes while its generator is loaded with the emergency loads.

6. Verifying the diesel generator operates for $\geq$ 60 minutes while loaded to $\geq$ 2750 kw.

7. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 3000 kw.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 3.2% $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN required by Specification 3.1.1.1 is based upon this limiting condition and is consistent with FSAR accident analysis assumptions. For earlier periods during the fuel cycle, this value is conservative. With $T_{avg} \leq 200°F$, the reactivity transients resulting from any postulated accident are minimal and a 1% $\Delta k/k$ shutdown margin provides adequate protection.

3/4.1.1.3 BORON DILUTION AND ADDITION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration changes in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 10,060 + 700/-0 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron concentration changes will be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurance that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

The MTC is expected to be slightly negative at operating conditions. However, at the beginning of the fuel cycle, the MTC may be slightly positive at operating conditions and since it will become more positive at lower temperatures, this specification is provided to restrict reactor operation when T_{avg} is significantly below the normal operating temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 1.0% Δk/k after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 4550 gallons of 6.25% boric acid solution from the boric acid tanks or 47,300 gallons of 1720 ppm borated water from the refueling water storage tank.

The requirements for a minimum contained volume of 370,000 gallons of borated water in the refueling water storage tank ensures the capability for borating the RCS to the desired level. The specified quantity of borated water is consistent with the ECCS requirements of Specification 3.5.4. Therefore, the larger volume of borated water is specified here too.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

BASES

3/4.5.1 SAFETY INJECTION TANKS

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

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

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

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

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

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provides this protection by dissolving in the sump water and causing its final pH to be raised to $\geq$ 7.0.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the accident analyses are met and that subsystem OPERABILITY is maintained. The requirement to dissolve a representative sample of TSP in a sample of RWST water provides assurance that the stored TSP will dissolve in borated water at the postulated post LOCA temperatures.

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

3/4.5.4 REFUELING WATER STORAGE TANK (RWST)

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

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.

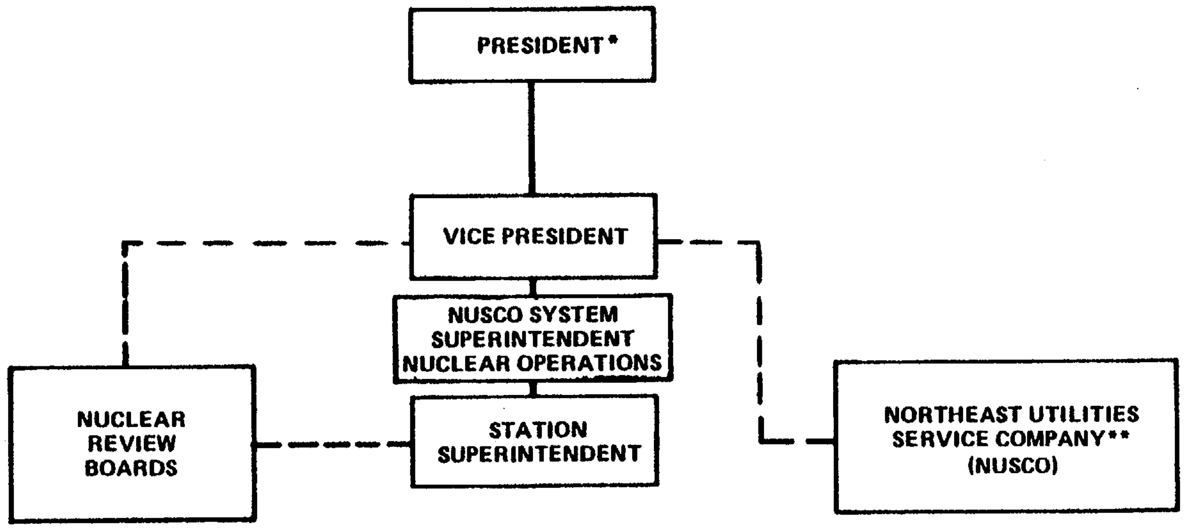
e. All CORE ALTERATIONS after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

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

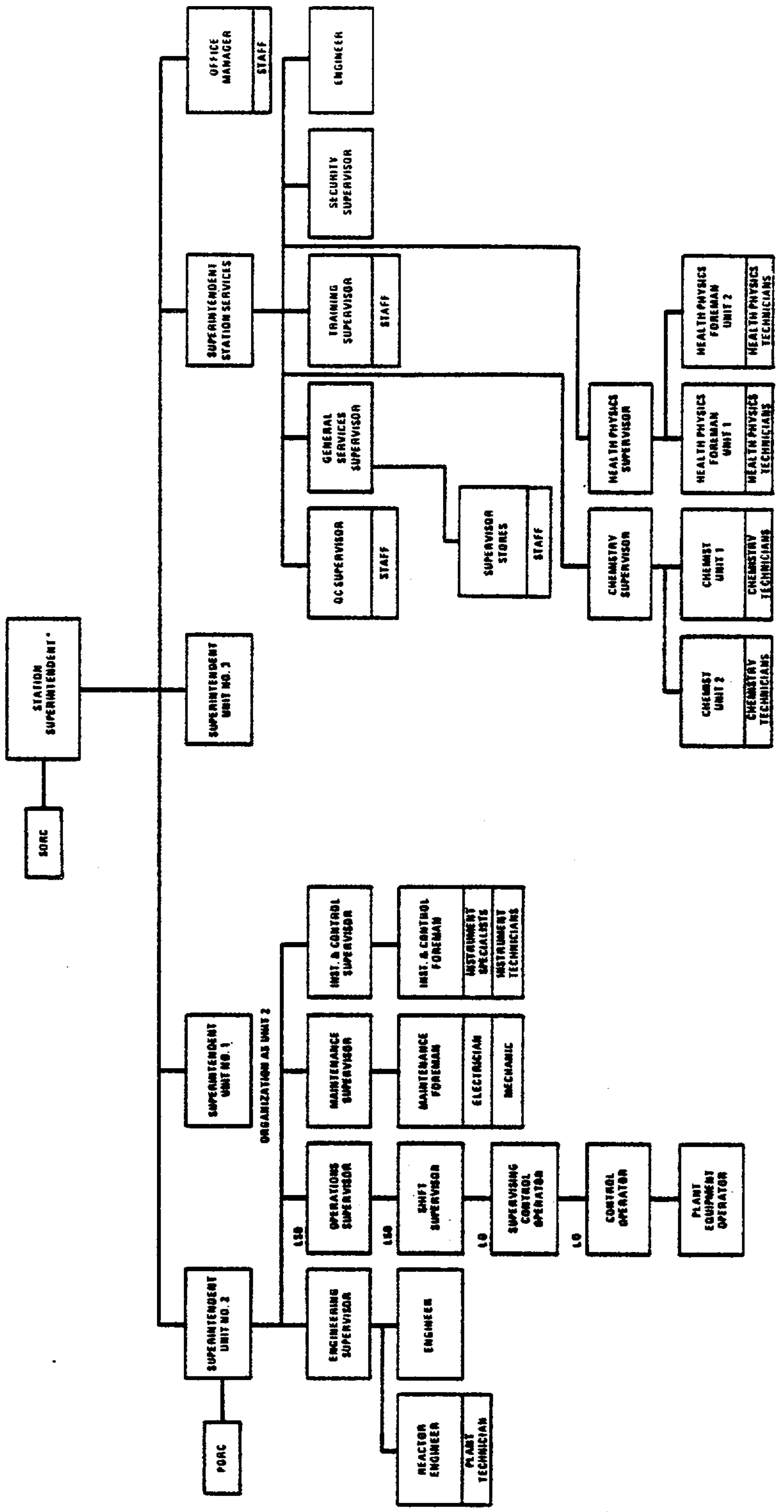
6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, Revision 1, after January 1, 1978.

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



* OVERALL SITE RESPONSIBILITY FOR FIRE PROTECTION

CODE: LO - LICENSED OPERATOR
LSO - LICENSED SENIOR OPERATOR

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

*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.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Station Superintendent and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Training Supervisor and shall meet or exceed the requirements of Section 27 of the NFPA Code-1976, except for Fire Brigade training sessions which shall be held at least quarterly.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC)

FUNCTION

6.5.1.1 The PORC shall function to advise the Unit Superintendent on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PORC shall be composed of the:

Chairman:	Unit Superintendent
Vice Chairman & Member:	Operations Supervisor
Member:	Maintenance Supervisor
Member:	Instrument and Control Supervisor
Member:	Reactor Engineer
Member:	Senior Engineer or Startup Supervisor*
Member:	Chemistry Supervisor or Station Services Superintendent of Service Group Supervisor or Health Physics Supervisor
Member:	Staff Engineer**

ALTERNATES

6.5.1.3 Alternate members shall be appointed in writing by the PORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate in PORC activities at any one time.

*When position is staffed.

**The Staff Engineer member of the PORC shall have an academic degree in engineering of physical science field; and in addition, shall have a minimum of five years technical experience, of which a minimum of three years shall be in the nuclear power plant industry.

MEETING FREQUENCY

6.5.1.4 The PORC shall meet at least once per calendar month and as convened by the PORC Chairman.

QUORUM

6.5.1.5 A quorum of the PORC shall consist of the Chairman or Vice Chairman or Station Superintendent and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The PORC shall be responsible for:

a. Review of 1) all procedures, except common site procedures, required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Unit Superintendent to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to Sections 1.0 - 5.0 of these Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications and preparation and forwarding of a report covering evaluation and recommendations to prevent recurrence to the System Superintendent Nuclear Operations and to the Chairman of the Nuclear Review Board.

f. Review of events requiring 24 hour notification to the Commission.

g. Review of facility operations to detect potential safety hazards.

h. Performance of special reviews and investigations and reports thereon as requested by the Chairman of the Nuclear Review Board.

i. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a) through (e) above constitutes an unreviewed safety question.

AUTHORITY

6.5.1.7 The PORC shall:

a. Recommend to the Unit Superintendent written approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

b. Provide immediate written notification to the Station Superintendent, System Superintendent Nuclear Operations and the Nuclear Review Board of disagreement between the PORC and the Unit Superintendent; however, the Unit Superintendent shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The PORC shall maintain written minutes of each meeting and copies shall be provided to the Station Superintendent, System Superintendent Nuclear Operations and Chairman of the Nuclear Review Board.

6.5.2 SITE OPERATIONS REVIEW COMMITTEE (SORC)

FUNCTION

6.5.2.1 The SORC shall function to advise the Station Superintendent on all matters related to nuclear safety of the entire Millstone Station Site.

COMPOSITION

6.5.2.2 The SORC shall be composed of the:

Chairman:	Station Superintendent
Member:	Unit 1 Superintendent
Member:	Unit 2 Superintendent
Member:	Unit 3 Superintendent
Member:	Designated Member of Unit 1 PORC
Member:	Designated Member of Unit 2 PORC
Member:	Designated Member of Unit 3 PORC
Member:	Station Services Superintendent

ALTERNATES

6.5.2.3 Alternate members shall be appointed in writing by the SORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate in SORC activities at one time.

MEETING FREQUENCY

6.5.2.4 The SORC shall meet at least once per six months and as convened by the SORC Chairman.

QUORUM

6.5.2.5 A quorum of the SORC shall consist of the Chairman and four members including alternates.

RESPONSIBILITIES

6.5.2.6 The SORC shall be responsible for:

a. Review of 1) all common site procedures required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Station Superintendent to affect site nuclear safety.

b. Review of all proposed changes to Section 6.0 "Administrative Controls" of these Technical Specifications.

c. Performance of special reviews and investigations and reports as requested by the Chairman of the Site Nuclear Review Board.

d. Review of the Plant Security Plan and implementing procedures and shall submit changes to the Chairman of the Site Nuclear Review Board.

e. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the Site Nuclear Review Board.

f. Review of all common site proposed tests and experiments that affect nuclear safety.

g. Review of all common site proposed changes or modifications to systems or equipment that affect nuclear safety.

RESPONSIBILITIES (Continued)

h. Render determinations in writing or meeting minutes with regard to whether or not each item considered under 6.5.2.6(a) through (g) above constitutes an unreviewed safety question.

AUTHORITY

6.5.2.7 The SORC shall:

a. Recommend to the Station Superintendent written approval or disapproval in meeting minutes of items considered under 6.5.2.6(a) through (g) above.

b. Provide immediate written notification or meeting minutes to the System Superintendent Nuclear Operations and the Site Nuclear Review Board of disagreement between the SORC and the Station Superintendent; however, the Station Superintendent shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.2.8 The SORC shall maintain written minutes of each meeting and copies shall be provided to the System Superintendent Nuclear Operations and Chairman of the Site Nuclear Review Board.

6.5.3 NUCLEAR REVIEW BOARD (NRB)

FUNCTION

6.5.3.1 The NRB shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

COMPOSITION

6.5.3.2 The Chairman, members and alternate members of the NRB shall be appointed in writing by the Vice President, Engineering and Operations, and shall have an academic degree in an engineering or physical science field; and, in addition, shall have a minimum of five years technical experience, of which a minimum of three years shall be in their respective fields of expertise. No more than two alternates shall participate as voting members in NRB activities at any one time.

CONSULTANTS

6.5.3.3 Consultants shall be utilized as determined by the NRB Chairman to provide expert advice to the NRB.

MEETING FREQUENCY

6.5.3.4 The NRB shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.3.5 The minimum quorum of the NRB necessary for the performance of the NRB review and audit functions of these technical specifications shall consist of the Chairman or his designated alternate and at least four members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW

6.5.3.6 The NRB shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes in Sections 1.0 - 5.0 of these Technical Specifications or licenses.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. REPORTABLE OCCURRENCES requiring 24 hour notification to the Commission.

h. Indications of a significant unanticipated deficiency, affecting nuclear safety, in some aspect of design or operation of safety related structures, systems or components.

i. Reports and meetings minutes of the PORC.

AUDITS

6.5.3.7 Audits of facility activities shall be performed under the cognizance of the NRB. These audits shall encompass:

a. The conformance of facility operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per year.

AUDITS (Continued)

b. The performance, training and qualifications of the entire facility staff at least once per year.

c. The results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per six months.

d. Any other area of facility operation considered appropriate by the NRB or the Vice President Nuclear Engineering and Operations.

AUTHORITY

6.5.3.8 The NRB shall report to and advise the Vice President Nuclear Engineering and Operations on those areas of responsibility specified in Sections 6.5.3.6 and 6.5.3.7.

RECORDS

6.5.3.9 Records of NRB activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each NRB meeting shall be prepared, approved and forwarded to the Vice President Nuclear Engineering and Operations within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.3.6 above, shall be prepared, approved and forwarded to the Vice President Nuclear Engineering and Operations within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.3.7 above, shall be forwarded to the Vice President Nuclear Engineering and Operations and to the management positions responsible for the areas audited within 30 days after completion of the audit.

AUTHORITY

6.5.4.9 The SNRB report to and advise the Vice President Nuclear Engineering and Operations on those areas of responsibility specified in Sections 6.5.4.7 and 6.5.4.8.

RECORDS

6.5.4.10 Records of SNRB activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each SNRB meeting shall be prepared, approved and forwarded to the Vice President Nuclear Engineering and Operations within 14 days following each meeting.

b. Reports of reviews encompassed by, Section 6.5.4.7 above, shall be prepared, approved and forwarded to the Vice President Nuclear Engineering and Operations within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.4.8 above, shall be forwarded to the Vice President Nuclear Engineering and Operations and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE Report requiring 24 hour notification to the Commission shall be reviewed by the PORC and submitted to the NRB and the System Superintendent Nuclear Operations.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The provisions of 10 CFR 50.36(c)(1)(i) shall be complied with immediately.

SAFETY LIMIT VIOLATION (Continued)

b. The Safety Limit violation shall be reported to the Commission, the System Superintendent Nuclear Operations and to the NRB immediately.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the NRB and the System Superintendent Nuclear Operations within 10 days of the violations.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972.

b. Refueling operations.

c. Surveillance activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed by the PORC/SORC, as applicable, and approved by the Unit Superintendent/Station Superintendent prior to implementation and reviewed periodically as set forth in each document.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the PORC/SORC, as applicable, and approved by the Unit Superintendent/Station Superintendent within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS[1/]

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions,[2/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

b. The complete results of steam generator tube inservice inspections performed during the report period (reference Specification 4.4.5.5.b).

1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

2/ This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

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.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NOS. 56 AND 45
PROVISIONAL OPERATING LICENSE NO. DPR-21 AND
FACILITY OPERATING LICENSE NO. DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL

MILLSTONE NUCLEAR POWER STATION, UNITS NOS. 1 AND 2

DOCKETS NOS. 50-245 AND 50-336

1.0 Introduction

By applications dated July 21, October 4, 12 and 18, November 14, 16 and 21, and December 13 and 15, 1977; and January 12 and 24, February 23, and March 20 and 21, 1978, Northeast Nuclear Energy Company (NNECO or the licensee) requested amendments to Provisional Operating License No. DPR-21 and Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Units Nos. 1 and 2.

The proposed changes to the Technical Specifications (TS) for Unit 2 only consist of:

- changing the refueling water storage tank sump recirculation actuation setpoint;
- revising the engineered safety features response times;
- modifying the incore detector operability requirements to be more definite and to remove unnecessary requirements;
- correcting the required number of redundant meteorological monitoring instruments;
- modifying the action requirements for operability of the control room chlorine detectors;
- defining "immediate" in certain ACTION statements and revising other specified time intervals;
- correcting the pressure at which the safety injection tank isolation valves must be operable;
- increasing the amount of TSP required to neutralize the containment sump after a LOCA;

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- changing the wording to indicate that specific doorways in the enclosure building have only one door;
- requiring verification of proper operation of the diesel generator under simulated emergency conditions; and
- adding surveillance requirements for ECCS throttled valves.

The proposed changes to the TS for both Units Nos. 1 and 2 consist of:

- changing administrative controls to reflect current organizational structure; and
- providing greater flexibility regarding entry into high radiation areas.

2.0 Discussion and Evaluation

2.1 RWST Level Transmitter

While filling the refueling pool for the last Millstone Unit 2 refueling outage from the Refueling Water Storage Tank (RWST), the licensee noted that even though tank level was reduced well below the Sump Recirculation Actuation Signal (SRAS) setpoint, initiation did not occur. Investigation of the instrumentation malfunction indicated that each channel trip was offset from the SRAS setpoint. The problem was caused by trapped air in the level transmitter piping configuration.

The RWST level transmitters were replaced with delta-P transmitters which have the capability of being completely vented. The licensee has documented that the replacement delta-P transmitters satisfy qualification requirements of the applicable standards as identified in Section 7.1.2 "Identification of Safety Criteria" of the FSAR.

In addition, NNECO has proposed a change to TS Table 3.3-4 for Unit 2 which provides Engineered Safety Feature Actuation System instrumentation trip values.(1) The licensee has proposed to change the level setpoint from 30 inches above the tank bottom to a range of setpoints from 2.5 feet, the current setpoint, to 5.5 feet above the tank bottom. The lower limit is sufficiently above the lowest level which can be detected by the new transmitters. Therefore measurement and system SRAS initiation at this level is assumed. The upper limit ensures that Specification 3.5.4.a, which requires a minimum contained water inventory of 370,000 gallons in the RWST, is satisfied. It has been determined that the volume of water between the 5.5 foot level and the RWST Low Level Alarm (34'-7") is greater than 376,000 gallons, adequately fulfilling the requirements of Specification 3.5.4.a. These proposed changes to the TS provide the added flexibility by allowing adjustments in both the SRAS setpoint and the RWST Low Level Alarm as may be desirable to accommodate changes in RWST inventory.

By inclusion of these proposed changes, there is continued assurance that a minimum contained volume of 370,000 gallons in the RWST will be available, and that a timely switchover to the sump recirculation mode will be accomplished when required. Therefore, we conclude that the proposed change is acceptable.

2.2 Engineered Safety Features Response Times

The licensee has proposed changes to TS Table 3.3-5 for Unit 2 that provide separate response times for the high pressure safety injection (HPSI) and low pressure safety injection (LPSI) systems and revised response times for the containment spray (CS) system and the enclosure building filtration system (EBFS).[2,3]

2.2.1 HPSI and LPSI Response Times

NNECO has proposed to reduce the response time for safety injection flow from the HPSI and LPSI pumps with offsite power available from 30.0 to 5.0 seconds. The acceleration time for both pumps at rated voltage is 4.0 seconds, as shown in Table 6.3-2 of the FSAR. The safety analysis of a postulated steam line break assumes a HPSI and LPSI pump response time of 5.0 seconds. The proposed TS are consistent with that analysis. Although the present requirement is a 30.0 second response time, the HPSI pumps have demonstrated the capability of performing their safety function within a 5.0 second time limit.

They also propose to introduce a maximum response time of 50.0 seconds for LPSI flow when power from the diesel generators is required. This time interval includes the following conservative assumptions:

- A safety injection actuation signal (SIAS) is received one second after the rupture;
- The diesel generators are assumed to start 20.0 seconds after the SIAS;
- A delay of 14.6 seconds is allowed for the load sequencer (including uncertainties) to start the LPSI pump; and
- Eight seconds are allowed to accelerate the pump should only 70% of the rated voltage be available.

This sequence of events indicates that 43.6 seconds after the rupture, flow is available from the LPSI pumps. With offsite power unavailable, the large break LOCA is the limiting incident. In that analysis, however, no credit is taken for LPSI flow until the safety injection tanks are empty which is calculated to be approximately 60 seconds following the actuating signal. Therefore, a 50.0 second response time for the LPSI pumps can be accommodated by the installed equipment. This is reflected in the present accident analysis.

The changes to the response times for HPSI and LPSI are acceptable because they are in the conservative direction with respect to the FSAR and the values used in the previous cycle.

2.2.2 Containment Spray Response Time

NNECO has proposed to change the required response time for the CS from 34.5 to 35.6 seconds when the diesel generators are required. The CS is actuated by a high-high containment pressure of 27 psig which occurs 6 seconds after the rupture for the design basis incident. The diesel generators receive the signal to start one second after the rupture and are five seconds into their starting step when the Containment Spray Actuation Signal (CSAS) is received. This point in time is the $\tau = 0$ stage for the CS as defined by TS definition 1.26. An additional 15 seconds are allowed for the diesel generators to start in the worst case and 14.6 seconds more (including sequencer uncertainties) are allotted before the spray pumps are started. Allowing 6.0 seconds for the pumps to accelerate to full speed with 70% of the rated voltage, 35.6 seconds have elapsed since the initiating CSAS (41.6 seconds after the rupture). This summarizes the basis for the proposed 35.6 second specification. At this time, water is filling the CS headers. Time to fill those headers is not and has not previously been included in the response time and a footnote to that effect is also included in this proposed change for clarity. The reason this interval is not included is that surveillance testing of the headers is not practical. An interval of 14.3 seconds is needed for the pump to supply water to the CS nozzles. Therefore, 55.9 seconds after the break, water has reached the CS nozzles.

The change in containment spray response time is not in the conservative direction. However, the revised response time provides that flow from the spray nozzles is available at 55.9 seconds after the design basis break, whereas the FSAR value for the time that spray flow is available is 56.3 seconds after the DBE. Thus, the original conclusion of the FSAR remains valid and these changes are acceptable, and do not significantly reduce the margin.

2.2.3 Enclosure Building Filtration System Response Time

NNECO has proposed to increase the allowable response time for the Enclosure Building Filtration System (EBFS) to account for the actual time for the diesel generators to start and for the load sequencer to start the EBFS fans following a postulated loss-of-coolant accident and assuming a loss of offsite power.

In its safety evaluation of the design basis loss-of-coolant accident (SER dated May 10, 1974), the staff assumed that one minute was required for the EBFS to reduce the enclosure building pressure to a minimum of -0.25 inches of water (gauge) relative to outside pressure. The licensee has determined since then that the actual time required approaches two minutes due to time required to start the diesel generators, to sequence the EBFS load, and to drawdown the enclosure building air volume to the required negative pressure.

We have reviewed the licensee's analysis of the offsite consequences with the increased response time and the earlier analysis performed for the May 10, 1974 SER. The duration of unfiltered release will increase from one minute to two minutes. The effect of this increased response time, at worst, will be to double that portion of the total dose calculated in the 1974 SER that comes from the unfiltered leakage. This assumes that the previously reported X/Q values are still acceptable. The staff has reviewed more recent meteorological data from the Millstone site and has concluded the X/Q values in the May 10, 1974 SER are appropriately conservative. However, in our reanalysis, the 1974 SER X/Q values were used. Based on this, an increase in the EBFS response time from 35 seconds to 50 seconds and in the total time required to reach -0.25 inch water gauge in the enclosure building from one minute to 110 seconds would increase the two-hour exclusion area boundary dose as shown in the table below. The 10 CFR Part 100 limits and the original SER values are included for comparison.

LOSS-OF-COOLANT ACCIDENT DOSE (REMS)

Limit and Calculation Results	Site Boundary		Low Population Zone	
	Thyroid	Whole Body	Thyroid	Whole Body
10 CFR Part 100 Limit	300	25	300	25
May 10, 1974 SER	172	6.1	90	2.6
NNECO Analysis of 2/23/78	116	1.2	66	1.4
NRC Revised Evaluation	210	7	94	3

We conclude that this change to the EBFS response time TS (Table 3.5-5) remains acceptable and does not significantly decrease the margin to the 10 CFR Part 100 limits. In this connection it should be noted that:

- the X/Q values used in the NRC evaluation are known to be appropriately conservative;
- the NRC calculated dose values are well below the acceptable limits of 10 CFR Part 100; and
- the calculated values for LOCA consequences for Millstone Unit 2 remain less than the consequences computed for Millstone Unit 1. The Millstone Unit 1 computations remain controlling with respect to site boundary dose effects.

The emergency plan designed to protect the general public in the event of an accident remains unaffected by this modification.

2.2.4 Diesel Generator Response Time

In the above evaluation of emergency safeguard response times, a diesel generator response time of 20 seconds was used by the licensee. The Millstone Unit No. 2 Final Safety Analyses Report (FSAR) states that the diesel generators are capable of starting within a nominal 12 seconds. No TS limit is presently specified for this parameter. In response to a NRC request, NNECO has proposed a TS limit on diesel generator response time of 20 seconds (the value used in their analysis for the emergency safeguards system)(18). The licensee provided an additional response that documented:

- their intent to investigate and take appropriate corrective action if the diesel generator start time is in excess of the nominal 12 seconds used in the FSAR;
- their analysis of the response of all equipment powered by the diesel generators under loss of offsite power conditions, both permanently connected loads and loads added at each sequence step, and the conclusion that a 20 second start time is substantiated; and
- the diesel generator start times for both units since early 1976. A brief description of deficiencies, where available, was provided(19).

We conclude that a diesel generator response time of 20 seconds is acceptable to provide emergency safeguards electrical loads, under loss of offsite power conditions, as evaluated in Sections 2.2.1, 2.2.2 and 2.2.3 of this SE. This response time will be specified in TS Section 4.8.1.1.2. We further conclude that the automatic sequence time delay relays should be surveillance tested on a 18 month schedule. NNECO has agreed to this TS additions to Section 4.8.1.1.2.

2.3 Incore Detectors

NNECO proposed a change to TS Section 3.3.3.2 for Unit 2 which addresses the operability of the incore detector system.(4) The Incore Detector System is required for four types of monitoring in the CE reactors:

1. Recalibrations of the excore neutron flux detection system;

2. Monitoring the azimuthal power tilt;

3. Calibration of the power level neutron flux channels; and

4. Monitoring the linear heat rate.

The present TS specify one set of operability requirements to encompass all four functions. This set consists of two requirements:

- At least 75% of detector locations shall be OPERABLE.

- There shall be at least two sets of fourfold 90° symmetric OPERABLE detector locations.

An OPERABLE location was defined as a location in which at least three of the possible four segments are OPERABLE. Therefore, considerable usable data was discarded by this definition of OPERABLE.

The proposed change would delete the requirements that would prevent reactor start-up with an inoperable system or cause reactor shutdown within 30 hours following inoperability of the system. We have recently issued a new standard TS on this subject to Baltimore Gas and Electric Company (BG&E) for Calvert Cliffs Units Nos. 1 and 2. NNECO has modified their initial proposal to accept this new standardized TS for Millstone Unit No. 2.

In the proposed TS for Incore Detectors, the requirements for each function the incore system performs are treated separately, and thus unnecessary requirements can be relaxed. Conversely, for each function being performed by the incore system, the TS requirements are based on the actual mathematical model used to satisfy the requirement. The result is a more logical and consistent TS.

We have specified an absolute minimum requirement that at least one operable detector segment in each core quadrant on each of the four axial elevations be operable. The remainder of the operability requirements are as presented in the following discussions on each function of the Incore Detector System.

2.3.1 Monitoring Azimuthal Power Tilt

Since tilt values are computed on each detector level, the tilt computation on a given level does not require the operability of any detectors in that string except those on that level. The OPERABILITY requirement based on locations will be replaced by the requirement that sufficient detector segments be OPERABLE to compute at least two tilt values on each of the four detector segment levels. This change in Specification 3.3.3.2 greatly increases the number of tilt values that can be computed and is acceptable.

2.3.2 Recalibration of the Excore Neutron Flux Detector System

Again the proposed TS places requirements on OPERABLE segments rather than OPERABLE locations. The proposed TS requires that at least 75% of the detector segments remain OPERABLE. In addition it requires a minimum of nine OPERABLE incore detector segments on each detector level and a minimum of two OPERABLE detector segments in the inner 109 fuel assemblies and two OPERABLE segments in the outer 108 fuel assemblies. These new requirements guarantee that adequate instrumentation in each region of the core is available to monitor core performance.

The change from requiring 75% of detector locations be OPERABLE to requiring 75% of detector segments be OPERABLE is conservative. The criterion of requiring that 75% of detector segments be operable was demonstrated by a simulation study by BG&E. The simulation study consisted of first measuring the ASI with a full compliment of detector segements and comparing this with measurements of ASI with up to 25% of the detector segments failed. The selection of failed detectors was constrained so that there remained at least four OPERABLE segments on each detector level so that large regions of the core could not become uninstrumented. This restraint to four OPERABLE segments per level is conservative compared with the proposed TS requirements of nine OPERABLE segments per detector level and at least two OPERABLE segments in the inner 109 fuel assemblies and two OPERABLE segments in the outer 108 fuel assemblies per detector level. Thus, the simulation study represents a very conservative approach. The results of the study showed that with up to 25% failed detectors there is a negligible loss of resolution in the measurement of ASI. Therefore, we find the proposed TS acceptable.

2.3.3 Monitoring Peaking Factors and Linear Heat Rates

The present TS is based on having 75% of detector locations OPERABLE. The uncertainties applied to the setpoints are based on the concept of OPERABLE locations, rather than OPERABLE segments. Furthermore, the core follow program, which assures that the incore detector measurement errors are within the 3.95% bound assumed in the INCA Topical Report (CENPD 145), is also based on operable locations. Thus, this part of the proposed

TS continues to require 75% of the detector locations to be operable. The same two requirements of the previous section are also required to guarantee that no large region of the core becomes uninstrumented for this monitoring. These proposed TS requirements give better assurance than the current TS that improved monitoring is used. On this basis, we conclude that the proposed TS are acceptable.

2.4 Meteorological Monitoring Instrumentation

NNECO has proposed a change to TS Section 3.3.3.4 for Unit 2 which describes the meteorological monitoring equipment.[5] The existing TS requires three instruments, each, for indication of wind speed, wind direction and air temperature-delta T. For each indication, two of the three instruments are required to be operable. The proposed change would reduce the number of instruments, from three to two each, for the above described indications and require both to be operable. NNECO has stated in the application that only two instruments, rather than three, per indication were installed for Millstone Unit No. 2.

Regulatory Guide 1.23, entitled, "Onsite Meteorological Programs," states in Section C.5 that "Meteorological instruments should be inspected and serviced at a frequency which will assure at least a 90% data recovery and which will minimize extended periods of instrument outage. The use of redundant sensors and/or recorders may be another acceptable means of achieving the 90% data recovery goal." Our review of the October 12, 1977 application indicates that two instruments (both of which must be operable), each, for windspeed, wind direction and air temperature-delta T, meet the "redundancy" requirement of Regulatory Guide 1.23 in that at least a 90% data recovery rate is expected. Accordingly, the proposed change to TS 3.3.3.4 is acceptable.

2.5 Chlorine Detection System

NNECO has proposed a change in the Chlorine Detection System Technical Specification action statement for Unit 2 to remove the requirement for plant shutdown in the event that both control room chlorine detectors are declared inoperable.(6) They point out that inoperability of this system should require that the control room emergency ventilation system be maintained in the recirculation mode. In this mode, all outside air dampers are closed and outside air is not introduced into the system. NNECO concludes that this is the most conservative action that can be taken and that placing the reactor in the cold shutdown condition does not increase the level of operator protection. The staff agrees with this analysis.

In preparing a new Specification 3.3.3.6, we have expanded the action statements to cover single and both chlorine detection system failures. NNECO has agreed with this modification of their submittal.

2.6 Definition of "Immediate" in Certain Action Statements

NNECO has proposed to supply definite time restrictions for the required performance in place of the undefined "immediate" action and revise other specified time intervals to more reasonably reflect time requirements for certain plant actions for Unit 2.(7) The TS to be modified require appropriate response times when ACTION statements are exceeded to restore the condition to normal or take further action to remedy plant condition.

We find that the proposed changes provide appropriately defined times for important plant action, are in accordance with other Standard TS for CE designed facilities and, therefore, are acceptable.

2.7 Safety Injection Isolation Valves

The safety injection tanks at Millstone Unit No. 2 are provided with isolation valves to prevent discharge of the tanks when the reactor coolant system (RCS) pressure is intentionally lowered below the discharge pressure of the tanks.(8) These valves are designed to open automatically above a predesignated pressure of about 300 psia or in the event of a safety injection signal, to prevent the safety injection tanks from being defeated. TS Section 4.5.1.d requires that the safety injection tank isolation valves be demonstrated to open automatically, when the RCS pressure exceeds 300 psia on a safety injection signal, once per 18 months. NNECO has proposed a change that would require demonstration of safety injection tank isolation valve operability before pressure exceeds 1750 psia instead of the existing 300 psia requirement.

TS 3.5.1 requires the safety injection tanks to be operable above 1750 psia. The demonstration of safety injection tank isolation valve operability below 1750 psia does not materially add to reactor safety in that the safety injection tanks are not required to be operable in this range. Demonstrating operability of the isolation valves at a pressure in excess of

300 psia and below 1750 psia is appropriate since it is consistent with the operability requirements on the safety injection tank. Accordingly, the proposed change to Technical Specification 4.5.1.d is acceptable.

2.8 Minimum Requirements for Trisodium Phosphate Dodecahydrate (TSP)

In response to our Bulletin of November 4, 1977 in which we questioned the minimum requirements for TSP to neutralize the containment sump following a LOCA, NNECO proposed to revise Unit 2 TS Surveillance Requirement 4.5.2.C.3.(9)

TSP is added to the containment sump by dissolutions from baskets on the containment floor. The function of TSP in the Millstone Unit No. 2 facility is to raise the pH and prevent chloride stress corrosion of components inside containment, including the containment liner. Standard Review Plan Section 6.1.3 states that available information indicates optimum pH control consists of stabilizing pH between 7 and 8 within four hours after a postulated LOCA. The FSAR concludes that 65 cubic feet is a quantity sufficient to raise the sump water pH to a value greater than 7.0

NNECO has determined that the original calculations made to determine the amount of TSP required assumed minimum water volumes and boron concentrations. In their reanalysis, they assumed maximum water volumes and boron concentrations of 2400 ppm for the Refueling Water Storage Tank, Safety Injection Tanks, and the Reactor Coolant System (RCS) and 21,000 ppm for the Boric Acid tanks. This results in increasing the minimum TSP requirement from 65 cubic feet to slightly less than 110 cubic feet. The proposed TS (Section 4.5.2.C.3) would require 110 cubic feet. If the boron concentrations in the various tanks are assumed at TS minimum values and the RCS is assumed at 0 ppm boron, NNECO has calculated the resultant pH of the containment sump after a LOCA to be about 7.4.

We conclude that this proposed correction to the TS increasing the TSP requirement to 110 cubic feet meets our requirements and is, therefore, acceptable.

2.9 Enclosure Building Doors

Section 4.6.5.2 of the present TS for Unit 2 requires that, when Enclosure Building (EB) Integrity is required, at least one door shall be closed in each access opening when the opening is being used for normal transit entry and exit. NNECO proposed to modify the TS by removing the requirement to have one door closed since only one door exists at seven EB access openings.(10)

The FSAR describes the EB as a limited leakage steel frame structure completely surrounding the containment above grade. It is designed and constructed to ensure that an acceptable upper limit of leakage of radioactive materials to the environment would not be exceeded in the unlikely event of a LOCA. Amendment No. 39 to the FSAR presents the detailed design of the EB including the location of the seven single door access openings. In our May 10, 1974 SE, we concluded that the reactor containment, including the EB, was designed in accordance with our requirements given in GDC 16 and 50 and Appendix A to 10 CFR Part 50. Therefore, we conclude that removing the requirement to have one EB door closed during normal transit entry and exit is an acceptable correction to TS Section 4.6.5.2.

2.10 AC Emergency Power Supply

By letter dated June 2, 1977, we informed NNECO of our concern regarding the integrity of the Millstone Unit No. 2 emergency AC power capability in the event of a degraded grid voltage occurrence. In our letter, we requested NNECO to submit (1) a demonstration of compliance with Staff Positions regarding plant electrical performance under degraded grid voltage conditions, and (2) a submittal of proposed Technical Specifications to include an improved test of emergency AC power capability. NNECO chose to demonstrate compliance with the Staff Positions by presenting their plan in a submittal dated July 21, 1977.[12] The necessary TS were submitted by application dated October 18, 1977.[11]

The proposed TS provides for an additional test of emergency AC power capability. The test, described in proposed TS Section 4.8.1.1.2.b.3.d, conforms to guidance contained in our letter of June 2, 1977 in that it tests whether the Millstone Unit No. 2 emergency AC power capability will retain its integrity under postulated degraded grid voltage conditions. The proposed TS change is, therefore, acceptable.

2.11 ECCS Throttle Valves

HPSI and LPSI designs of many Pressurized Water Reactors (PWR) utilize a common low pressure and a common high pressure header to feed the several cold (and in some cases hot) leg injection points. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration; (2) provide a proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. On many plants, there are motor operated valve(s) in the lines to each injection point that have stops which are set during pre-operational flow testing of the plant to insure that these flow requirements are satisfied. On other plants, electrical or mechanical stops on the Safety Injection System's isolation valve(s) are used for this purpose. Millstone 2 utilizes the former to satisfy these ECCS flow requirements.

While pre-operational HPSI/LPSI flow testing is utilized to assure that the valves used to throttle flow have been properly set, we have concluded that periodic surveillance requirements are needed to assure that these settings are maintained throughout the life of the plant. Consequently, we requested all PWR licensees to propose changes to their TS, as appropriate, to incorporate periodic surveillance requirements for these valves. Sample surveillance requirements, developed by the NRC staff, were provided to licensees for guidance in developing proposed changes.

The sample requirements include periodic verification of throttle valve position stop settings, and verification of proper ECCS flow rates whenever system modifications are made that could alter flow characteristics. The request for proposed TS changes was sent to NNECO on June 30, 1977.

NNECO responded to our request with respect to Millstone Unit No. 2 by proposing changes to the TS Section that are in essential agreement with the staff's requirements.(13) Based on our review, we have concluded that the licensee's proposed increased surveillance requirements would provide sufficient additional assurance that proper valve settings for ECCS flows and flow distributions will be maintained throughout the plant life; and thus, the proposed changes are acceptable.

2.12 Administrative Controls

NNECO proposed to revise TS Section 6 for both units to reflect current organizational structure.(14) We have been informed of several minor changes since the original submittal, which have been incorporated. Since these are purely administrative changes, they involve no safety considerations and are, therefore, acceptable.

2.13 High Radiation Area

NNECO proposed to revise TS Section 6.13.1 for both units to provide greater flexibility regarding entry into high radiation areas.(15) The present TS requires that any individual or group of individuals permitted to enter high radiation areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area. The proposed TS would also authorize entry under continuous Health Physics coverage or with use of an integrating dosemeter with preset alarm capabilities. We proposed that NNECO accept the more recent version of the Standard TS that provides the flexibility that NNECO proposed, but contains more precise requirements that we find necessary. NNECO has agreed to this new version of the STS. We conclude that this change, provides radiation protection for employees equivalent to the present TS, is consistant with that authorized for other STS facilities and is, therefore, acceptable.

3.0 Environmental Consideration

We have determined that the amendments do 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 amendments involve 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 these amendments.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: December 8, 1978

REFERENCES

1. NNECO application to revise the containment sump recirculation actuation signal (SRAS) setpoint, D. Switzer to G. Lear, January 24, 1978.

2. NNECO application for revised engineered safety features response times, D. Switzer to G. Lear, December 13, 1978.

3. NNECO application to modify the enclosure building filtration system response time, D. Switzer to R. Reid, February 23,1978.

4. NNECO application on incore detector operability requirements, D. Switzer to G. Lear, November 16, 1977.

5. NNECO application for meteorological instrumentation system change, D. Switzer to G. Lear, October 12, 1977.

6. NNECO application to revise the chlorine detection system action statement, D. Switzer to G. Lear, October 4, 1977.

7. NNECO application to supply definite time restrictions replacing "immediate" actions, D. Switzer to G. Lear, November 14, 1977.

8. NNECO application on safety injection isolation valves, D. Switzer to G. Lear, October 18, 1977.

9. NNECO application on minimum required amount of TSP, D. Switzer to G. Lear, December 15, 1977.

10. NNECO application on enclosure building design, D. Switzer to G. Lear, November 21, 1977.

11. NNECO application on AC emergency power supply, D. Switzer to G. Lear, October 18, 1977.

12. NNECO submittal demonstrating compliance with Staff Positions on AC emergency power supply, D. Switzer to G. Lear, July 21, 1977.

13. NNECO application on ECCS throttle valves surveillance requirements, D. Switzer to G. Lear, January 12, 1978.

14. NNECO application to change administrative controls to reflect current plant organizational structure, D. Switzer to D. Ziemann and R. Reid, March 21, 1978.

15. NNECO application on entry into high radiation areas, D. Switzer to D. Ziemann and R. Reid, March 20, 1978.

16. NNECO application on containment leak rate test intervals, D. Switzer to R. Reid, March 14, 1978.

17. NNECO withdrawal of application on containment leak rate test intervals, W. Counsil to R. Reid, July 31, 1978.

18. NNECO application to incorporate diesel generator response time in TS, W. Counsil to R. Reid, August 11, 1978.

19. NNECO additional information on diesel generator response time, W. Counsil to R. Reid, October 11, 1978.

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-245 AND 50-336

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL

NOTICE OF ISSUANCE OF AMENDMENTS TO OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 56 and 45 to Provisional Operating License No. DPR-21 and Facility Operating License No. DPR-65, respectively, to Northeast Nuclear Energy Company, The Connecticut Light and Power Company, The Hartford Electric Light Company, and Western Massachusetts Electric Company, which revised Technical Specifications for operation of the Millstone Nuclear Power Station, Units Nos. 1 and 2, located in the Town of Waterford, Connecticut. The amendments are effective as of their date of issuance.

The amendments revise the Appendix A Technical Specifications for Unit No. 2 only by:

- changing the refueling water storage tank sump recirculation actuation setpoint;
- revising the engineered safety features response times;
- modifying the incore detector operability requirements to be more definitive and to remove unnecessary requirements;
- correcting the required number of redundant meteorological monitoring instruments;
- modifying the action requirements for operability of the control room chlorine detectors;
- defining "immediate" in certain ACTION statements and revising other specified time intervals;

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- correcting the pressure at which the safety injection tank isolation valves must be operable;
- increasing the amount of TSP required to neutralize the containment sump;
- changing the wording to indicate that specific doorways in the enclosure building have only one door;
- requiring verification of proper operation of the diesel generator under simulated emergency conditions; and
- adding surveillance requirements for ECCS throttled valves.

The amendments revise the Appendix A Technical Specifications for both Units 1 and 2 by:

- changing administrative controls to reflect current organizational structure; and
- providing greater flexibility regarding entry into high radiation areas.

The applications for the amendments comply 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 amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments 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 these amendments.

For further details with respect to this action, see (1) the applications for amendments dated July 21, October 4, 12, and 18, November 14, 16, and 21, and December 13, and 15, 1977, and January 12 and 24, February 23, and March 20 and 21, 1978, (2) Amendments Nos. 56 and 45 to Licenses Nos. DPR-21 and DPR-65, respectively, 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, Route 156, 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 Operating Reactors.

Dated at Bethesda, Maryland, this 8th day of December 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors