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Docket No. 50-336

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

Dear Mr. Counsil:

The Commission has issued the enclosed Amendment No. 50 to Facility Operating License No. DPR-65 for Millstone Nuclear Power Station, Unit No. 2. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated December 8, 1977, and supplemental information dated December 3, 1976, March 8 and 22 and June 9, 1977.

The amendment modifies the TS to incorporate low temperature overpressure protection system requirements. Some of your proposed TS changes have been modified to meet our requirements. These modifications have been discussed with and accepted by your staff.

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

Sincerely,

Original signed by

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

Enclosures:
1. Amendment No. 50
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

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OELD
Amend. & FR Notice only. See comment on p.2 of FR Notice JRG 3/21/79

*See previous yellow for concurrence

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2/13/79

BGrimes
2/14/79

OFFICE	ORB#3:DOR	ORB#4:DOR	ORB#1:DOR	OELD	C-ORB#4:DOR	
SURNAME	SSHEPPARD	MCONNER/cb	GZECH		RREID	
DATE	2/2/79	2/8/79	2/12/79	2/ /79	2/ /79	

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

March 23, 1979

Docket No. 50-336

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Northeast Nuclear Energy Company
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cc w/enclosures:
See next page

Northeast Nuclear Energy Company

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

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

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

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

Chief, Energy Systems Analyses
Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, N.W.
Washington, D. C. 20460

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

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

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

U. S. Nuclear Regulatory Commission
Region I
Office of Inspection and Enforcement
ATTN: Mr. John T. Shedlosky
631 Park Avenue
King of Prussia, Pennsylvania 19406

cc w/enclosure(s) and incoming
dtd.: 12/3/76, 3/8,22/77,
& 12/8/77
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. 50-336

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 50
License No. DPR-65

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

   A. The application 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 December 8, 1977, and supplemental information dated December 3, 1976, March 8 and 22 and June 9, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

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

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

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

7905070337

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

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 23, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 50

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

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

Pages

V
3/4 4-1
3/4 4-21a (added)
3/4 4-21b (added)
3/4 5-7
B 3/4 4-1
B 3/4 4-2
B 3/4 4-2a
B 3/4 4-11
B 3/4 4-12
6-21

INDEX

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1 Four reactor coolant pumps shall be in operation.

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

ACTION:

MODES 1 and 2:

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

MODES 3, 4** and 5**:

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

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exception 3.10.4.

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

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

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 4 and 5.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.2 The pressurizer code safety valve shall be demonstrated OPERABLE per Surveillance Reguirement 4.4.3.

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.

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two power operated relief valves (PORVs) with a lift setting of ≤ 450 psig, or

b. A reactor coolant system vent of ≥ 1.3 square inches.

APPLICABILITY: When the temperature of one or more of the RCS cold legs is ≤ 275°F, except when the reactor vessel head is removed.

ACTION:

a. With one PORV inoperable, either restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through a ≥ 1.3 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

b. With both PORVs inoperable, depressurize and vent the RCS through a ≥ 1.3 square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

c. In the event either the PORVs or the RCS vent(s) are used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE.

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months.

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

d. Testing in accordance with the inservice test requirements for ASME Category C valves pursuant to Subsection IWV-3510 of Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition, and Addenda through Summer 1975.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

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.

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

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

4.5.3.2 All high-pressure safety injection pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is $\leq$ 275°F by verifying that the motor circuit breakers have been removed from their electrical power supply circuits.

---

*With pressurizer pressure < 1750 psia.

#A maximum of one high-pressure safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is $\leq$ 275°F.

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.

3/4.4.1 REACTOR COOLANT LOOPS

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

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

The restrictions on starting a Reactor Coolant Pump during MODES 4 and 5 with one or more RCS cold legs $\leq$ 275°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 43°F (31°F when measured by a surface contact instrument) above the coolant temperature in the reactor vessel.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

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

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

3/4.4.4 PRESSURIZER

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

3/4.4.5 STEAM GENERATORS

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

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

Stress corrosion cracking could also be initiated from the primary side, if sufficiently large tube strains were introduced as a result of the primary and secondary effects of denting. The Supplementary Inspection Program assures that tubes that have developed excessive strains will be identified and removed from service, on a preventive basis, before cracking would develop, in accordance with the Tube Plugging Criteria described in Specification 4.4.5.2. Futhermore, the potential causative factors for developing excessive tube strain have been eliminated, or greatly reduced by (a) the steam generator repairs that were implemented during the November 1977 Outage, (b) condenser integrity resulting from the condenser retubing implemented in the May 1977 Outage, and (c) the phasing in of the Full Flow Condensate Polishing System during Cycle 2.

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

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

Mechanical tube defects caused by loose parts are unlikely, based on experimental data addressing loose parts effects. To provide conclusive assurance of the validity of this statement, and to demonstrate that hypothesized degradation does not occur, suspect tubes are to be inspected during the "Scheduled Inspection" as defined in Specification 4.4.5.2.4.

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

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

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

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

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

3/4.4.10 STRUCTURAL INTEGRITY

The required inspection programs for the Reactor Coolant System components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent practicable, the inspection program for the Reactor Coolant System components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code "Inservice Inspection of Nuclear Reactor Coolant Systems" dated July 1, 1971.

All areas scheduled for volumetric examination have been pre-service examined using equipment, techniques and procedures anticipated for use during post-operation examinations. To assure that consideration is given to the use of new or improved inspection equipment, techniques and procedures, the Inservice Inspection Program will be periodically reviewed on a 5 year basis.

The use of conventional nondestructive, direct visual and remote visual test techniques can be applied to the inspection of most reactor coolant loop components except the reactor vessel. The reactor vessel requires special consideration because of the radiation levels.

The techniques anticipated for inservice inspection include visual inspections, ultrasonic, radiographic, magnetic particle and dye penetrant testing of selected parts.

The nondestructive testing for repairs on components greater than 4 inches diameter gives a high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. Repairs on components 4 inches in diameter or smaller receive a surface examination which assures a similar standard of integrity. In each case, the leak test will ensure leak tightness during normal operation.

For normal opening and reclosing, the structural integrity of the Reactor Coolant System is unchanged. Therefore, satisfactory performance of a system leak test at 2250 psia following each opening and subsequent reclosing is acceptable demonstration of the system's structural integrity. These leak tests will be conducted within the pressure-temperature limitations for Inservice Leak and Hydrostatic Testing of Specification 3.4.9.1 and Figure 3.4-2.

Inspection of the pipe hangers and supports provides assurance that these devices are operated within permissible travel and/or loading limits.

3/4.4.11 CORE BARREL MOVEMENT

This specification is provided to ensure early detection of excessive core barrel movement if it should occur. Neutron noise levels are used to continually monitor core support barrel (CSB) motion. Change in motion is manifested as changes in the four excore neutron detector signals. Baseline core barrel movement Alert Levels and Action Levels at nominal THERMAL POWER levels of 20%, 50%, 80% and 100% of RATED THERMAL POWER will be determined during the reactor startup test program.

Data from these detectors is to be reduced in two forms. RMS values are computed from the Amplitude Probability Density (APD) of the signal amplitude. These RMS magnitudes include variations due both to various neutronic effects and internals motion. Consequently, these signals alone can only provide a gross measure of CSB motion. A more accurate assessment of CSB motion is obtained from the Auto and Cross Power Spectral Densities (PSD, XPSD), phase (ø) and coherence (COH) of these signals. These data result from a Spectral Analysis (SA) of the excore detector signals.

A modification to the required monitoring program may be justified by an analysis of the data obtained and by an examination of the affected parts during the plant shutdown at the end of the first fuel cycle.

THIRTY-DAY WRITTEN REPORTS (Continued)

completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety features instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety features systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.8.c, above, designed to contain radioactive material resulting from the fission process.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.
b. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.
c. Safety Class 1 Inservice Inspection Program Review, Specification 4.4.10.1.
d. Core Barrel Movement, Specifications 3.4.11 and 4.4.11.
e. ECCS Actuation, Specifications 3.5.2 and 3.5.3.
f. Fire Detection Instrumentation, Specification 3.3.3.7.
g. Fire Suppression Systems, Specifications 3.7.9.1 and 3.7.9.2.
h. RCS Overpressure Mitigation, Specification 3.4.9.3.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

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

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

c. All REPORTABLE OCCURRENCES submitted to the Commission.

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

e. Records of reactor tests and experiments.

f. Records of changes made to operating procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

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

6.10.2 The following records shall be retained for the duration of the facility operating license:

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

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

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

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

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

1.0 Introduction

By application dated December 8, 1977, and supplemental information dated December 3, 1976, March 8 and 22 and June 9, 1977, Northeast Nuclear Energy Company, et al, (NNECO or the licensee) requested changes to the Technical Specifications (TS) for the Millstone Nuclear Power Station, Unit No. 2.

The proposed changes to the TS consist of adding low temperature overpressure protection system (OPS) requirements.

2.0 Background

The history of the generic low temperature overpressure protection issue is described in NUREG-0138 (Reference 1). Briefly, a series of over 30 incidents had occurred in pressurized water reactors (PWRs) since 1972 in which the Appendix G pressure-temperature limits had been exceeded at temperatures less than normal operating temperature.

These incidents consisted of two varieties of pressure transients: a mass input type from charging pumps, safety injection pumps, or safety injection accumulators, and an energy input type caused by thermal feedback when a reactor coolant pump (RCP) sweeps cooler primary system water through a steam generator with a hot secondary side. These incidents usually occurred in a water solid system during startup or shutdown operations.

Pressure transients which could occur at normal operating temperature, approximately 570 F, are mitigated in most plants by large code safety valves located on the pressurizer. These are mechanical valves which open against a spring pressure of about 2400 psia. The code safety valves are quite simple, having no electrical components, and as such are considered passive, failure free components. These code safety valves are tested in accordance with ASME Code, Section XI requirements.

Prior to the introduction of an OPS, pressure transients initiated while operating at lower temperatures were not protected against and there were no pressure relief devices in the reactor coolant system to prevent these transients from exceeding the Appendix G pressure-temperature limits. Nuclear reactors such as Millstone 2, which have a pressure limit in excess of 2500 psia at 570 F, have only a 700 psia limit at 200 F. The code safety valves with settings in the 2400 psia range would not be able to relieve a pressure transient at low RCS temperature without the Appendix G limits being violated by a large amount.

The Appendix G pressure limit drops off rapidly at lower temperatures because the reactor vessel material and welds have significantly less toughness at lower temperatures and are therefore more susceptible to flaw induced failure. In addition, factors such as copper content in welds and neutron fluence levels affect the material toughness and contribute to the reduction in safety margin to vessel failure at low temperature conditions. The Millstone 2 overpressure protection analysis was performed utilizing the Appendix G curves for 2 to 10 years of full power operation as the basis for maximum allowable pressure.

As a solution to the low temperature overpressurization problem, the licensee identified a set of power operated relief valves (PORV's) located on the pressurizer which are normally available for overpressure protection during normal plant operations. These usually have a single pressure setpoint just below the opening pressure of the mechanical code safety valves and are designed to relieve small pressure transients without requiring the code safety valves to lift. The licensee proposed to provide the PORV's with a low pressure setpoint to which they could be switched as the plant cooled down. If a pressure transient would occur at these lower temperatures and the lower setpoint had been selected, there would then be a pathway to relieve system pressure.

The PORV's are significantly more complicated than the code safety valves since the PORV's require electrical circuitry to sense pressure, transmit a signal to the valve, and actuate the solenoid to open the valve. Thus

it is desirable to insure redundancy and separability in the circuitry to preclude a single failure from disabling the entire OPS system.

In a series of meetings and through correspondence with PWR vendors and licensees, the staff developed a set of criteria, which if adhered to, would produce an acceptable OPS. These criteria are:

1. Operator Action: The licensee could not take credit for operator action for 10 minutes after the operator became aware of an ongoing transient.

2. Single Failure: The system had to be designed to relieve overpressure transients assuming the worst case single failure in addition to the event which caused the transient.

3. Testability: The system had to be testable on a periodic basis consistent with the system's employment.

4. Seismic and IEEE 279 Design: Ideally the system should meet seismic Class I and IEEE 279 design requirements. The basic objective is that the system should not be vulnerable to a common failure mode which both initiated a pressure transient and caused a failure of equipment needed to terminate the transient.

In addition to the four formally stated criteria mentioned above, a number of additional criteria were established in the process of the staff review of generic submittals from the various vendors and in the exchange of information between the staff and the licensees.

Foremost among these was the requirement that the licensees show protection for the limiting mass addition transient regardless of the administrative procedures proposed to eliminate that potential scenario. Each licensee, therefore, was required to analyze the effects of the single pump start which would produce the most limiting mass addition transient and most severely challenge the Appendix G limits. For Millstone 2 a High Pressure Safety Injection (HPSI) pump start produces the most limiting pressure transient.

For the worst case energy addition transient the licensees were allowed to limit the severity of the transient in their analyses by assuming a maximum ΔT across the steam generator. By maximum ΔT we mean the maximum difference in the temperature between the primary loop coolant and the secondary loop water in the steam generator. For this case and for other scenarios the licensees were required to develop Technical Specifications which delineated the actions required to limit the severity of these scenarios and also provide justification for their action.

Another criterion for the design of the OPS was that the electrical instrumentation and control system provide a variety of alarms to alert the operator to 1) properly enable the low temperature OPS at the proper temperature during cooldown, and 2) indicate if a pressure transient was occurring. Additionally the electrical system had to provide positive assurance that the isolation valve upstream of each PORV was open when the system was enabled by wiring its position into the enable alarm. The enable alarm would not be permitted to clear until the OPS mode selector switch for each PORV system was placed in the low pressure setpoint position and the isolation valve was opened.

NNECO submitted a generic overpressurization protection report prepared by Combustion Engineering (CE) in Reference 2. This report was prepared for the CE Owner's Group comprising five utilities. The generic report provided information on RCS response to postulated pressure transients that occur at low temperatures during heatup and cooldown, and provided a general description of design modifications which could be used to prevent overpressurization of CE designed Nuclear Steam Supply Systems (NSSS). The staff, in conjunction with its review of the CE generic report, requested that NNECO commit to a schedule for implementing a permanent or interim version of the OPS by December 31, 1977, and requested additional information related to the application of the generic aspect of the OPS as pertinent to the Millstone 2 plant (Reference 3). In References 4 and 5 the licensee submitted additional information to the staff on equipment and procedural improvements as well as a schedule for implementation of the proposed system.

NNECO states that the Millstone 2 final OPS was installed during the refueling outage beginning in December 1977. This conforms to the staff's requirement to install an interim or final version of OPS by December 31, 1977.

NNECO submitted the Millstone 2 plant specific report in Reference 7 and additional plant specific data was supplied in Reference 9.

3.0 Discussion and Evaluation

The system installed by NNECO for Millstone 2 incorporates a defense in depth concept for overpressure protection, utilizing operator training, administrative procedures, Technical Specifications, and hardware improvements to meet the criteria established by the staff. The objective of the OPS is, first, to insure that pressure transients while operating at low RCS temperatures become and remain unlikely events, and second, to mitigate the consequences of a pressure transient should one occur. The mitigating system includes sensors, actuating mechanisms, and valves to prevent a RCS pressure transient from exceeding the pressure-temperature limits included in the Millstone 2 Technical Specifications as required by Appendix G to Chapter 10, Code of Federal Regulations, Part 50 (10 CFR 50). These Appendix G limits are those established by using procedures defined in Appendix G to Section III of the ASME Code. Appendix G to 10 CFR 50 states that these ASME Code limits can be used for startup and shutdown when the reactor is not critical. For criticality, Appendix G to 10 CFR 50 requires more stringent rules than Appendix G to Section III of the ASME Code.

Suggested TS were submitted by the licensee as part of the OPS plan and are reviewed together with the system hardware in this Safety Evaluation. However, the format and content of the proposed TS needs to be revised to meet our requirements. NNECO has agreed to such modifications of the TS for OPS.

3.1 Technical Specifications and Operating Procedures

One cornerstone of the Millstone 2 OPS is the use of TS and operating procedures to limit the probability of initiating pressure transients at low temperatures (<275 F) and to insure the enabling, disabling, and proper functioning of the OPS.

The TS specify the conditions required for starting a RCP, the PORV OPERABILITY requirements and the PORV surveillance requirements. We conclude that these TS will provide assurance that pressure transients at low temperatures will be unlikely and that the system will function to prevent overpressure transients from exceeding Appendix G limits. We further conclude that the TS meet the criteria established by the staff and are acceptable.

The licensee will make extensive use of operating procedures to provide a large measure of the administrative protection against overpressure transients. Among these operating procedures for low temperature operating conditions are the following:

- When RCS temperature, pressure, and other operating conditions permit, a pressurizer steam volume of 60% of the pressurizer volume will be maintained.

- The TS require the maximum ΔT across the steam generator to be less than 31 F prior to starting a RCP at low temperature. This insures that functioning of one of the two PORV's will provide sufficient relief capacity such that Appendix G will not be violated in the event of an inadvertent RCP start.

- Emergency Core Cooling System (ECCS) component testing will be conducted with a steam bubble or with the reactor vessel head removed. Operational testing of the Safety Injection and Chemical and Volume Control System (CVCS) components (i.e., pumps, valves, automatic signals, etc.) will be accomplished with a non-solid RCS.

We conclude that these operating procedures contribute measurably to plant protection from low temperature overpressure transients.

The steam generator ΔT of 31 F specified in the TS as the operating limit is a result of determining the minimum ΔT required to prevent an Appendix G violation and then factoring in uncertainties. In this particular case the maximum permissable ΔT, calculated to be 43 F, was reduced by uncertainties in instrument accuracy (9 F) and the difference in steam generator bulk fluid and shell side temperature (3 F). We conclude that the licensee's method of measuring steam generator ΔT is acceptable.

We conclude that the TS submitted by the licensee will provide assurance that pressure transients at low temperatures will be unlikely and that the system will function to prevent overpressure transients from exceeding Appendix G limits. We further conclude that the TS meet the criteria established by the staff and are, therefore, acceptable.

3.2 Hardware

3.2.1. OPS Functioning

Acceptable performance of the OPS depends on the proper functioning and adequate relief capacity of the two PORV's located on the pressurizer. The NSSS vendor and the licensee demonstrated that with two PORV's functioning, all postulated mass and energy addition transients could be mitigated. If one PORV is assumed to fail, administrative procedures must be relied upon to limit the severity of the limiting transients in both the energy and mass addition cases to insure that Appendix G limits are not violated.

3.2.2. Energy Addition Transients

Administrative procedures, backed by TS require that the maximum ΔT across the steam generator be less than 43 F to lessen the consequences of the RCP start energy addition transient. As previously noted, the uncertainty due to instrument inaccuracy reduces the maximum ΔT to 40 F. This ΔT will be incorporated into the Millstone 2 TS in order to limit the severity of the design base energy addition transient. The operating procedures will actually call for the ΔT across the steam generator to be limited to no more than 5 F during a cooldown. This will provide additional assurance that a single PORV will be able to function to relieve this transient and maintain RCS pressure below Appendix G limits. Numerous assumptions were employed in the modeling of PORV relief to insure conservatism in the analysis of this design base energy addition transient:

- The RCS was assumed to be water solid.
- The RCS was also assumed to be rigid during the transient (no expansion).
- A single PORV was assumed to fail.
- RCS letdown flow was assumed isolated.
- Heat absorption by the RCS metal mass was not considered.

- Conservatively high heat transfer coefficients were utilized across the steam generator.

- RCP start was assumed to be instantaneous.

With a PORV low pressure setpoint of 465 psia the licensee showed that one PORV will provide sufficient relief capacity to give a maximum pressure of ~500 psia for an energy addition transient. This peak pressure corresponds to an Appendix G limit that would exist at temperatures well below the refueling temperature (~130 F). This analysis assumed a ΔT of no more than 43 F across the steam generator. As previously noted the licensee will monitor the temperature difference to insure a ΔT of no more than 40 F which in turn insures a ΔT of no more than 43 F when the 3 F of uncertainties are factored in.

We conclude that the licensee and vendor have demonstrated that the OPS can protect the RCS from exceeding Appendix G limits for an energy addition transient even with the additional single failure of a PORV.

3.2.3. Mass Addition Transients

Protection from the effects of the limiting mass addition transient was afforded by the licensee by assuring that components of the ECCS system would be disabled by procedure and TS during cooldown. This is accomplished at 250 F by disabling one HPSI pump and by disabling the second HPSI pump at 190 F. Disabling is accomplished by racking down the HPSI breakers and closing the HPSI discharge or header isolation valves with the breakers in the OFF position. This provides assurance that Appendix G limits will not be violated should a single PORV fail prior to or during a mass addition transient. Conservatisms included in the limiting mass addition transient model, the inadvertent single HPSI pump start, were as follows:

- The RCS was assumed to be water solid.

- Letdown flow from the RCS was assumed isolated.

- The RCS was considered to be rigid during the transient (no expansion).

- Mass addition was assumed to occur with the highest fluid density that could occur. PORV relief was assumed to occur with the lowest fluid density that could occur.

- A single PORV was assumed to fail.

- A conservative Bernoulli equation was utilized to model PORV relief.

The staff guidance to the licensee for analyzing the mass addition transient was to show that Appendix G limits were not violated assuming that the safety injection pump which could produce the worst case transient inadvertently started, regardless of administrative procedures calling for disabling the pumps at various stages. For the Millstone 2 plant the worst pump start would be a HPSI pump. The licensee demonstrated that a single HPSI pump plus one charging pump mass input transient would produce a peak pressure of 465 psia, the PORV opening setpoint. The equilibrium pressure for the HPSI pump and one charging pump output balanced by single PORV relief is 460 psia. The quick opening time of the valve (~10 milliseconds) results in the transient immediately being relieved, producing the equilibrium pressure of 460 psia. This corresponds to an Appendix G limit that would exist at temperatures well below the refueling temperature (~130 F).

We conclude that the licensee has demonstrated that the OPS will prevent overpressurization of the RCS due to mass addition transients, assuming the single failure of a PORV.

3.2.4. Conclusion on OPS Hardware

The system presented by NNECO to provide protection for the Millstone 2 plant from low temperature overpressure transients provides assurance that these transients will be unlikely events and that, should they occur, the plant will be protected.

We conclude, therefore, that the Millstone 2 OPS meets the criteria established by the staff for overpressure protection and is, therefore, acceptable.

3.3 Electrical Instrumentation and Control

The NNECO design for the OPS is based on the use of existing PORV's located on the pressurizer to provide low temperature pressure relief capability. These valves may be operated manually by closing a switch or may be activated by a preset pressure and temperature signal to provide automatic pressure relief. The PORV's are of the electromagnetic solenoid operated type.

Existing reactor coolant system (RCS) temperature signals are used to perform four functions. Each function is implemented in a dual redundant mode utilizing a separate temperature system for each PORV.

The functions that the RCS temperature signals provide are: (1) energization and de-engerization of the PORV solenoids, (2) activation of overpressure transient alarms, (3) activation of "low pressure setpoint" alarms and (4) activation of "high pressure setpoint" alarms. The RCS pressure signal is not utilized in the activation of the "high pressure setpoint" alarm.

Assurance for preventing inadvertent blowdown at RCS temperatures above 275 F is provided by the inclusion of two separate hand switches, one for each PORV. A hand switch provides the following functions to its associated PORV valve system: (1) low setpoint permissive signal, (2) high setpoint permissive signal, and (3) isolation valve open/close signal.

In addition, the low pressure setpoint signals also form part of the dual-redundant "reset-to-low" and "reset-to-high" circuitry, that informs the operator to which position a hand-operated switch should be set. To ensure that the PORV low setpoint is enabled at the required RCS pressure and temperature by the operator, operator action and warning alarms have been incorporated into the modified PORV circuitry logic.

By normal plant cooldown procedures, both the RCS pressure and temperature are decreased uniformly down to 300 F and 400-500 psig. Prior to cooling the RCS below 275 F, normal operating procedures will require the operator to manually enable the PORV "Low" setpoint by resetting the hand switch to the "Low" position.

During plant heatup, normal operating procedures will maintain the RCS pressure below 400 psig until the RCS temperature is greater than 275 F. When the RCS temperature exceeds 275 F, normal operating procedures will require that the PORV's are reset to the "Hi" setpoint of 2,385 psig, normal plant heatup will continue accordingly.

3.3.1 Design Basis Criteria and Staff Positions

Millstone Unit 2 was evaluated under the guidance of the four design basis criteria stated in Section 2.0 of this evaluation, with specific attention given to various pertinent staff positions resulting from these criteria. Thus, Sections 3.1.1 through 3.1.4 are concerned with the four design basis criteria, while Sections 3.1.5 through 3.1.11 provide more specific details concerning staff positions and their impact on the design and operation of Millstone Unit 2.

3.3.1.1 Operator Action

In each design basis transient analyzed, no credit for operator action was assumed until 10 minutes after the initiation of the RCS overpressurization transient and after the operator was made aware of the overpressure transient. Operator awareness of the overpressure transient will be derived by a low temperature overpressure transient alarm.

3.3.1.2 Single Failure

The overpressure protection system is designed to protect the reactor vessel given a single failure in addition to a failure that initiated the overpressure transient. Redundant pressure protection channels and relief valves are used to satisfy the single failure criterion. Redundant pressure (and temperature) sensors, bistables, two full capacity PORV's and independent power sources are provided for the long-term overpressure mitigating system. The long-term mitigating system meets the single failure criterion and is acceptable.

3.3.1.3 Testability

There are two aspects associated with the testability of the overpressure protection system (OPS). The first concerns the program for testing the PORV's for low pressure protection system operability, and has resulted in a staff position that "the control circuitry from pressure sensor to valve solenoid should be stroked during each refueling. Deviations from this criterion should be justified". Consequently, the testability program for the PORV's will be as follows:

a. Verification of upstream isolation valves functioning once per cold shutdown.
b. Performance of a Channel Functional test of the control circuitry from the pressure sensor to the valve solenoid to be conducted once per refueling outage.
c. Performance of a Channel Calibration of the pressurizer pressure sensors once per 18 months.

The second aspect of testability involves the plant tests during cold shutdown which could result in RCS overpressurization above the minimum operating limit curves. These tests are:

a. The integrated emergency core cooling system test,
b. The temperature loop calibrations for the letdown isolation detectors, and
c. The periodic surveillance tests of the charging pumps.

The following preventive measures have been instituted to prevent inadvertent RCS overpressurization:

a. Integrated ECCS system testing will be performed with either (1) a pressurizer steam bubble, or (2) with the reactor vessel head removed,
b. Temperature loop calibration will only be performed after the charging pumps are secured, and
c. Surveillance testing of charging pumps will be conducted during a cold shutdown with non-water solid conditions.

When the testing of components is required that might cause an RCS pressure rise above the minimum pressure/temperature limit curves, the OPS will be operational consistent with the RCS temperature considerations.

The OPS satisfies the staff testability criteria and is acceptable.

3.3.1.4 Seismic Design and IEEE Standard-279

IEEE Std-279 and Seismic Criteria were considered in the design of the overpressure protection system. The Millstone, Unit 2, plant satisfies criteria in the following manner.

The design criteria for the relief valves (PORV's) and the associated instrumentation and control hardware, which are the long-term mitigating systems for low temperature RCS overpressurization, are based on the existing applicable plant criteria and the following considerations:
a. The mitigating system is designed against single failure. The system is not vulnerable to a failure mode that would both initiate a pressure transient and disable the OPS.
b. Whenever protection system instrumentation and control hardware are used as part of the mitigating system, isolation criteria will be in accordance with the applicable plant criteria to which the subject unit has been designed and licensed.

The present PORV's (RC-402 and RC-404) are solenoid operated power relief valves. The subject valves were designed and manufactured in accordance with the ASME Boiler and Pressure Vessel Code, Section III (1968 Edition) and the applicable ASME code for Pumps and Valves (November 1968 Edition). The subject valves are classified as seismic Class I valves.

The long-term overpressurization mitigation system also satisfies the intent of IEEE-279 with regard to the channel separation, independence and redundancy of both the temperature and pressure sensors, along with their associated electronics alarms and PORV's. In addition, both channels have separate electrically independent power supplies (two each for pressure circuitry, two each for temperature), as do the pilot valves for each PORV.

3.3.1.5 High Pressure Alarm

The staff position is that a high pressure alarm be used during low RCS temperature operations as an effective means to alert the operator that a pressure transient is in progress.

A description of the alarm system design is provided below:
The alarm annunciates on the main control board when the reactor coolant system temperature is less than 275 F and the reactor coolant pressure is greater than 400 psig. The annunciator provides both visable and audible signals. Operator action is required to acknowledge the alarm. Redundant

sensors are provided by the existing low scale pressurizer channels and the wide range cold leg temperature detectors.

The pressure signals are powered from a vital power source while the temperature signals are powered from a non-vital power source.

NNECO has made modifications so both temperature and pressure signals are powered by redundant vital power sources since both of these signals are used in the RCS high pressure alarm and both PORV channels.

We find the above electrical power sources for the temperature and pressure signals to be acceptable.

3.3.1.6 Isolation Valve Alarm

The staff position is that the position of the upstream isolation valve should be wired into the overpressure protection alarm so that the alarm will not clear unless the system is enabled and the isolation valve is open. Means should be provided to insure proper alignment of the isolation valve during overpressure protection system operation. A description of the alarm system is provided below:

The upstream PORV isolation valves (2-RC-403 and 2-RC-405) are wired into the RCS OPS such that the hand switch enactment of the protection system will result in the opening of the isolation valves. An open-close indication for each isolation valve is provided on the main control board.

We find this modification to be acceptable.

3.3.1.7 Enable Alarm

The staff requires that an alarm be activated as part of the plant cooldown process to insure that the PORV "Low" setpoint is activated before the RCS temperature is equal to or less than 275 F. A description of the alarm system is provided below:

To insure that the PORV "Low" setpoint is activated, a PORV "Low" reset alarm is activated when the RCS temperature is equal to or less than 275 F. Once the PORV's are reset to the "Low" relief position, an annunciator window will remain lit to indicate the "Low" PORV mode of operation. The annunciator will remain in this mode until the PORV's are reset to the "Hi" position. The overpressure transient alarm becomes operational only at RCS temperatures below 275 F. Once the PORV's are reset to provide low temperature relief at 450 psig, plant cooldown can be resumed.

We find this modification to be acceptable.

3.3.1.8 Disable Alarm

The staff requires that an alarm be activated as part of the plant heatup process to insure that the PORV's are reset to the "Hi" setpoint when the RCS temperature is greater than 275 F. A description of the alarm system is provided below.

During plant heatup, normal operating procedures will maintain the RCS pres-

sure below 400 psig until the RCS temperature exceeds 275 F. When the RCS temperature exceeds 275 F, normal operating procedures require that the PORV's are reset to the "Hi" setpoint relief of 2,385 psig. At the same time, the overpressure transient alarm will be de-energized when the RCS temperature exceeds 275 F. In order to assure that the PORV's are reset to the "Hi" setpoint, an alarm will be activated when the RCS pressure exceeds 425 psig. After the PORV's are reset to the "Hi" setpoint of 2,385 psig, normal plant heatup will continue accordingly.

We find this modification to be acceptable.

3.3.1.9 PORV Open Alarm

The staff requires that an alarm be activated to alert the operator that a PORV is in the open position. Millstone Unit 2 does not presently have this alarm. The licensee will be required to install a PORV Open Alarm in both OPS channels.

3.3.1.10 Pressure Transient Reporting and Recording Requirements

The staff position is that pressure transients which cause the overpressure protection system to function, thereby indicating the occurrence of a serious pressure transient, constitute a 30-day reportable event. In addition, pressure and temperature recording instrumentation are required to provide a permanent record of the pressure transient. The response time of the P/T recorders shall be compatible with the transient rates of 100 psig per second.

Appropriate instrumentation and recording equipment exists at Millstone Unit No. 2 which will provide a continuous and permanent record over the full range of primary system pressure and temperature. The sensing and recording equipment will be in service during startup and shutdown operations as well as during long periods of cold shutdown operations.

We find this implementation to be acceptable.

3.3.1.11 Disabling of Non-Essential Components During Cold Shutdown

The staff position requires the de-energizing of SIS pumps and closure of SI header/discharge valves during cold shutdown operations.

A description of the disabling of non-essential components during cold shutdown follows:

3.3.1.11.1 Inadvertent SIS Activation

The plant cooldown procedure requires that the high pressure safety injection pumps be de-energized with the breakers in the rack down position prior to decreasing the reactor coolant system temperature below 190 F. In addition, it is required that the pump discharge valves or the header isolation valves be closed with the breakers in the OFF position. The surveillance procedures which specify the integrated emergency core cooling system tests will require that the reactor vessel head be removed or a bubble be formed in the pressurizer prior to Phasing in the HPSI pumps.

The safety injection tank isolation valves are closed when the RCS pres-

sure decreases below 1750 psig.

The only circumstance for which the HPSI pumps and the discharge valves will not be isolated and de-energized will occur during the planned integrated emergency core cooling system test. However, the potential for RCS overpressurization due to inadvertent SIS injection will be minimized, since either (1) a pressurizer bubble will be required, or (2) the reactor vessel head will be removed prior to phasing in the SIS equipment.

The HPSI pump breakers are located in the 4160 volt switchgear rooms. The flow control valve breakers are located on the upper three elevations of the auxiliary building. All pump and breaker controls are located on the main control boards in the control room.

We find this system to be acceptable.

3.3.1.11.2 Inadvertent SDC Isolation

In the shutdown cooling system, reactor coolant is circulated using the low-pressure safety injection pumps. The flow path from the pump discharge runs through the shutdown cooling normally locked closed valves SI-452 and SI-453, through the shutdown cooling heat exchangers, and through normally closed valve SI-657 to the low-pressure safety injection headers. The circulating fluid flows through the core and is returned from the reactor coolant system through the shutdown cooling nozzle in the loop No. 2 reactor vessel outlet pipe. The coolant is returned to the suction of the low pressure safety injection pumps through valves SI-651 and SI-652. These valves are interlocked to prevent opening when the reactor coolant system pressure exceeds the design pressure of the shutdown cooling system. Each valve is independently controlled by separate instrumentation channels. In the case of the subject unit, the shutdown cooling isolation valves (SI-651 and SI-652) are activated when the RCS pressure exceeds 300 psig. The isolation valves are motor-operated valves. The shutdown cooling system contains two relief valves in the suction loop to the low-pressure safety injection pumps. Relief valves SI-469 and SI-468 have relief capacities of 5 gpm and 155 gpm respectively. The setpoint pressure for SI 468 is 300 psig.

We find this system to be acceptable.

3.3.2 Conclusion on Electrical Instrumentation and Control

The design of the Millstone Unit 2 low temperature OPS in the areas of electrical, instrumentation and control (EI&C) is in accordance with those design criteria originally prescribed by the staff and later expanded during subsequent discussions with the exceptions noted previously.

We find the EI&C aspects of the modifications acceptable on the basis that: (1) the overpressure protection system complies with IEEE Std 279-1971, and seismic criteria as identified in Section 2.0; (2) the system is redundant and satisfies the single failure criterion; (3) the system is testable on a periodic basis, and (4) the recommended TS reduce the probability of overpressurization events to an acceptable level. The licensee has installed a PORV Open Alarm (Section 4.4) in each channnel and provided redundant Class IE vital power sources for the pressure and temperature signals (Section 4.5) since both of these two signals are used in the RCS high pressure alarm and both PORV channels.

3.4 System Testing

The Millstone 2 OPS is designed to be capable of being tested to ensure that the input signal to the control system is correctly transmitted to the valve-operating solenoid. A channel functional test of the instrumentation and control hardware will be conducted once during each refueling. Channel calibration of the pressurizer sensors will be performed once per 18 months. The valve will be tested in accordance with ASME code, Section XI, Subsection IWV.

3.5 Environmental Consideration

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

3.6 Conclusion

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

Date: March 23, 1979

REFERENCES

1. U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director, NRR to NRR Staff", NUREG-0138, November 1976.

2. NNECO submittal of "Generic Report Overpressure Protection for Operating CE NSSS", D. Switzer to G. Lear, December 3, 1976.

3. NRC Staff Position on Overpressure Protection, G. Lear to D. Switzer, January 12, 1977.

4. NRC Request for Additional Information, G. Lear to D. Switzer, February 2, 1977.

5. NNECO Response to Staff Request, D. Switzer to G. Lear, March 8, 1977.

6. NNECO Response to Staff Request, D. Switzer to G. Lear, March 22, 1977.

7. NNECO submittal of Specific Plant Report, D. Switzer to G. Lear, June 9, 1977.

8. NRC Determined Deficiencies, G. Lear to D. Switzer, July 11, 1977.

9. NNECO Application for Overpressure Protection and Response to Staff Request, D. Switzer to G. Lear, December 8, 1977.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-336

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 50 to Facility Operating License No. DPR-65, issued to Northeast Nuclear Energy Company, The Connecticut Light and Power Company, The Hartford Electric Light Company, and Western Massachusetts Electric Company, (the licensees), which revised Technical Specifications for operation of the Millstone Nuclear Power Station, Unit No. 2 (the facility) located in the Town of Waterford, Connecticut. The amendment is effective as of its date of issuance.

The amendment modifies the existing Technical Specifications by incorporating low temperature overpressure protection system requirements.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

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Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the application for amendment dated December 8, 1977, and supplemental information dated December 3, 1976, March 8 and 22 and June 9, 1977, (2) Amendment No. 50 to License No. DPR-65, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. and at the Waterford Public Library, Rope Ferry Road, 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 23rd day of March 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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