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Recovery Officer - Technical Services
Northeast Nuclear Energy Company
c/o Ms. Patricia A. Loftus
Director - Regulatory Affairs
P. O. Box 128
Waterford, Connecticut 06385

November 1998

SUBJECT: MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2 - REVISION TO TECHNICAL SPECIFICATIONS BASES

Dear Mr. Bowling:

By letter dated July 30, 1998, Northeast Nuclear Energy Company (NNECO) provided the NRC with changes to Technical Specification (TS) Bases Sections 3/4.9.1, 3/4.1.1.3, 3/4.7.1.6, 3/4.7.7, 3/4.5.4, and 3/4.3.3.10. NNECO provided the revised TS Bases to the NRC for information only.

As you are aware, the TS bases are not part of the TSs as defined in 10 CFR 50.36. Changes to the TS Bases may be voluntarily made in accordance with the provisions of 10 CFR 50.59. Should the proposed change involve an unreviewed safety question pursuant to 10 CFR 50.59(a)(2), or involve a change in the interpretation of the implementation of the TSs (i.e., constitute a TS change), then the proposed change is to be provided to the NRC staff pursuant to the provisions of 10 CFR 50.59(c) and 10 CFR 50.90 for prior NRC review and approval.

The TS Bases provided by you are hereby returned to you and should be inserted in the TSs to ensure that the NRC and NNECO have identical TS Bases pages. The NRC staff did not perform an evaluation of the TS Bases revisions and NRC concurrence with the revisions is not implied by this letter. The NRC staff may review NNECO's evaluations that support these TS Bases revisions during the next inspection of the Millstone Nuclear Power Station, Unit No. 2's, implementation of 10 CFR 50.59.

Sincerely,

Stephen Dembek, Project Manager
Millstone Project Directorate
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosure: As stated

cc w/encl: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 24, 1998

Mr. Martin L. Bowling, Jr.
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Millstone Project Directorate
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REVISED TECHNICAL SPECIFICATION BASES

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

Replace the following pages of the Appendix A, Technical Specifications, with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
B3/4 9-1	B3/4 9-1
B3/4 1-1	B3/4 1-1
B3/4 7-3	B3/4 7-3
B3/4 7-5	B3/4 7-5
B3/4 5-2a	B3/4 5-2a
B3/4 3-5	B3/4 3-5

Enclosure

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) sufficient boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. Reactivity control in the water volume having direct access to the reactor vessel is achieved by determining boron concentration in the refueling canal. The refueling canal is defined as the entire length of pool stretching from refuel pool through transfer canal to spent fuel pool.

For the Cycle 13 mid-cycle core offload activities, the boron concentration of the water volumes in the steam generators and connecting piping may be as low as 1300 ppm. During REFUELING and/or CORE ALTERATIONS, the water volumes in the steam generators and connecting piping are stagnant and do not readily mix with the water in the reactor vessel. The water volumes in the pressurizer and connecting piping, shutdown cooling system (including reactor vessel and connecting piping), and refueling pool shall be maintained greater than 1950 ppm.

A boron dilution analysis has been performed which accounts for dilution of the shutdown cooling system with the water volumes from the steam generators and connecting piping. This analysis demonstrates that, in the unlikely event in which all of the water in the steam generators and connecting piping mixes with the water in the shutdown cooling system, the resulting shutdown cooling system boron concentration will remain greater than the required refueling boron concentration.

The surveillance requirement to verify that the boron concentration in the steam generators is greater than 1300 ppm prior to entering MODE 6 is consistent with the assumptions of the boron dilution calculation. The sample points are only located on the cold leg side of the steam generators. These sample points are representative of the water volumes in the steam generators (both hot and cold legs) and their connecting piping, based on the fact that uniform mixing of these water volumes at a boron concentration of approximately 1320 ppm had occurred prior to shutting off the reactor coolant pumps. In March 1996, the reactor coolant system was drained and subsequently refilled with water having a boron concentration greater than or equal to 1320 ppm. The boron concentration of the water in the steam generators and connecting piping is greater than 1300 ppm.

3/4.9.2 INSTRUMENTATION

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

3/4.9.3 DECAY TIME

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

3/4.1.1 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, the minimum SHUTDOWN MARGIN specified in the CORE OPERATING LIMITS REPORT 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 the reduced SHUTDOWN MARGIN specified in the CORE OPERATING LIMITS REPORT provides adequate protection.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 1000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during reductions in Reactor Coolant System boron concentration. This was done to prevent vortexing in the SDCS when in mid-loop operation, while being consistent with boron dilution analysis assumptions. A flow rate of at least 1000 GPM will circulate the full Reactor Coolant System volume in approximately 90 minutes. With the RCS in mid-loop operation, the Reactor Coolant System volume will circulate in approximately 25 minutes. The reactivity change rate associated with reductions in Reactor Coolant System boron concentration 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.

Revised by NRC letter dated November 24, 1998

3/4.7.1.4 ACTIVITY (Continued)

of 10 CFR Part 100 limits in the event of a steam line rupture. The dose calculations for an assumed steam line rupture include the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

3/4.7.1.6 MAIN FEEDWATER ISOLATION COMPONENTS (MFICs)

Feedwater isolation response time ensures a rapid isolation of feed flow to the steam generators via the feedwater regulating valves, feedwater bypass valves, and as backup, feed pump discharge valves. The response time includes signal generation time and valve stroke. Feed line block valves also receive a feedwater isolation signal since the steam line break accident analysis credits them in prevention of feed line volume flashing in some cases. Feedwater pumps are assumed to trip immediately with an MSI signal.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200-psig are based on a steam generator RT_{NDT} of 50°F and are sufficient to prevent brittle fracture.

3/4.7.3 REACTOR BUILDING CLOSED COOLING WATER SYSTEM

The OPERABILITY of the reactor building closed cooling water system ensures that sufficient cooling capacity is available for continued operation of vital components and Engineered Safety Feature equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

Revised by NRC letter dated November 24, 1998

3/4.7.7 SEALED SOURCE CONTAMINATION

The limitations on sealed source removable contamination ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the source material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) limits for plutonium. Leakage of sources excluded from the requirements of this specification represent less than one maximum permissible body burden for total body irradiation if the source material is inhaled or ingested.

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shield mechanism.

3/4.7.8 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Plant Operations Review Committee. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guide 8.8 and 8.10. The addition or deletion of any hydraulic or mechanic snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

prohibit a MODE change in this situation, this exemption will allow Millstone Unit No. 2 to enter MODE 4, take the steps necessary to make the HPSI pump capable of injecting into the RCS, and then declare the pump OPERABLE. If it is necessary to use this exemption during plant heatup, the appropriate action statement of Specification 3.5.3 should be entered as soon as MODE 4 is reached.

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) after a LOCA the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes. Small break LOCAs assume that all control rods are inserted, except for the control element assembly (CEA) of highest worth, which remains withdrawn from the core. Large break LOCAs assume that all CEAs remain withdrawn from the core.

Revised by NRC letter dated November 24, 1998

3/4.3.3.9 Radioactive Liquid Effluent Instrumentation

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with approved methods in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50. Monitoring of the turbine building sumps and condensate polishing facility floor drains is not required due to relatively low concentrations of radioactivity possible.

3/4.3.3.10 Radioactive Gaseous Effluent Instrumentation

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with approved methods in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to CFR Part 50.

Two types of radioactive gaseous effluent monitoring instrumentation, monitors and samplers, are being used at MP2 stack and MP1 main stack. Monitors have alarm/trip setpoints and are demonstrated operable by performing one or more of the following operations: CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST. Samplers are strictly collection devices made of canisters and filters. The CHANNEL CHECK surveillance requirements are met through (1) documented observation of the in-service rad monitor sample flow prior to filter replacement; (2) documented replacement of in-line iodine and particulate filters; and (3) documented observation of sample flow following the sampler return to service. The flow indicator is the only indication available for comparison. These observations adequately provide assurance that the sampler is operating and is capable of performing its design function.

There are a number of gaseous release points which could exhibit very low concentrations of radioactivity. For all of these release paths, dose consequences would be insignificant due to the intermittent nature of the release and/or the extremely low concentrations of radioactivity. Since it is not cost-beneficial (nor in many cases practical due to the nature of the release (steam) or the impossibility of detecting such low levels), to monitor these pathways, it has been determined that these release paths require no monitoring nor sampling.

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