September 14, 2004

Mr. David A. Christian
Senior Vice President and Chief Nuclear Officer
Dominion Nuclear Connecticut, Inc.
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 3 - ISSUANCE OF AMENDMENT RE: LIMITING SAFETY SYSTEM SETTINGS AND INSTRUMENTATION (TAC NO. MB6166)

Dear Mr. Christian:

The Commission has issued the enclosed Amendment No. 220 to Facility Operating License No. NPF-49 for the Millstone Power Station, Unit No. 3, in response to your application dated August 7, 2002, as supplemented by letter dated November 5, 2003.

The amendment revised the Technical Specifications (TSs) related to safety system settings. Specifically, the proposed changes revised: (1) TS 1.0 “Definitions;” (2) TS 2.2.1 “Limiting Safety System Settings - Reactor Trip System Instrumentation Setpoints;” (3) TS 3.3.1 “Reactor Trip System Instrumentation;” (4) TS 3.3.2 “Engineered Safety Features Actuation System Instrumentation;” (5) TS 3.7.7 “Control Room Emergency Ventilation System;” and (6) TS 3.8.3.1 “Onsite Power Distribution - Operating.” In addition, the appropriate TS Bases were revised to conform with the proposed changes.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/RA/

Victor Nerses, Senior Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-423

Enclosures: 1. Amendment No. 220 to NPF-49
2. Safety Evaluation

cc w/encls: See next page

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DOMINION NUCLEAR CONNECTICUT, INC., ET AL.

DOCKET NO. 50-423

MILLSTONE POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 220
License No. NPF-49

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

   A. The application for amendment by the applicant dated August 7, 2002, as supplemented November 5, 2003, 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.

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. NPF-49 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 220, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. Dominion Nuclear Connecticut, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of issuance, and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Daniel S. Collins, Acting Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 14, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 220

FACILITY OPERATING LICENSE NO. NPF-49

DOCKET NO. 50-423

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

Remove	Insert
1-3	1-3
2-7	2-7
3/4 3-3	3/4 3-3
3/4 3-10	3/4 3-10
3/4 3-13	3/4 3/13
3/4 3-22	3/4 3-22
3/4 3-25a	3/4 3-25a
3/4 7-17	3/4 7-17
3/4 8-16	3/4 8-16
3/4 8-17	3/4 8-17
B 2-8	B 2-8

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 220

TO FACILITY OPERATING LICENSE NO. NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.

MILLSTONE POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

By letter dated August 7, 2002, as supplemented by letter dated November 5, 2003, Dominion Nuclear Connecticut, Inc. (licensee or DNC), submitted a request for changes to the Millstone Power Station, Unit No. 3 (MP3) Technical Specifications (TSs). The requested changes would revise the TSs related to safety system settings. Specifically, the proposed changes would revise: (1) TS 1.0 "Definitions;" (2) TS 2.2.1 "Limiting Safety System Settings - Reactor Trip System Instrumentation Setpoints;" (3) TS 3.3.1 "Reactor Trip System Instrumentation;" (4) TS 3.3.2 "Engineered Safety Features Actuation System Instrumentation;" (5) TS 3.7.7 "Control Room Emergency Ventilation System;" and (6) TS 3.8.3.1 "Onsite Power Distribution - Operating." In addition, the appropriate TS Bases would be revised to conform with the proposed changes.

The November 5, 2003, letter provided clarifying information that did not change the scope of the amendment as described in the original *Federal Register* notice dated October 15, 2003 (67 FR 63687), and did not change the Nuclear Regulatory Commission (NRC or the Commission) staff's initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

Section 182a of the Atomic Energy Act of 1954, as amended, requires applicants for nuclear power plant operating licenses to include the TSs as part of the license. The Commission's regulatory requirements related to the content of the TSs are set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36(c). The regulation requires that the TSs include items in eight specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; (5) administrative controls; (6) decommissioning; (7) initial notification; and (8) written reports. However, the regulation does not specify the particular requirements to be included in a plant's TSs.

For LCOs, 10 CFR 50.36(c)(2)(ii) specifies four criteria to be used in determining whether a particular matter is required to be included in an LCO, as follows: (1) installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary (RCPB); (2) a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident (DBA) or transient analysis that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier; (3) a structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier; or (4) an SSC which operating experience or probabilistic safety assessment has shown to be significant to public health and safety. LCOs and related requirements that fall within or satisfy any of the criteria in the regulation must be retained in the TSs, while those requirements that do not fall within or satisfy these criteria may be relocated to licensee-controlled documents.

With regards to criterion (3) above, one such component that is part of the primary success path and which functions to mitigate DBAs is the safety injection (SI) input from the Engineered Safety Feature Actuation System (ESFAS). Upon receipt of a signal, the ESFAS initiates a reactor trip through the SI input from the ESFAS. However, other transients and accidents take credit for varying levels of engineered safety feature (ESF) performance and rely upon rod insertion to ensure reactor shutdown. Therefore, a reactor trip is initiated every time an SI signal is present.

Since the SI input from the ESFAS meets the requirements for inclusion in the TSs, its deletion from TS Table 2.2.1 should only be allowed if the removal does not affect its operability or testing requirements. Additionally, these requirements must be captured elsewhere in the TSs.

Furthermore, 10 CFR 50.36, “Technical Specifications,” requires that the TSs contain “...settings for automatic protective devices... so chosen that automatic protective action will correct the abnormal situation before a Safety Limit is exceeded.” One system that performs an automatic protection action is the reactor trip system (RTS). The RTS initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and reactor coolant system pressure limits during anticipated operational occurrences and to assist the ESF systems in mitigating accidents.

One input into the RTS is the SI input from the ESFAS. The automatic actuation logic of this function ensures that a reactor trip will occur upon receipt of any signal that initiates an SI. In order to comply with the above requirement of 10 CFR 50.36, any TS change to the setpoints related to the SI input from the ESFAS would have to continue to ensure that plant operation will not violate the safety limits.

General Design Criterion (GDC) 17, “Electric Power System,” of Appendix A, “General Design Criterion for Nuclear Power Plants,” to 10 CFR Part 50 requires that nuclear power plants have an onsite electric power system and an offsite electric power system to permit the functioning of SSCs important to safety. The safety function of each system (assuming the other system is not functioning) is to provide sufficient capacity and capability to assure that (1) fuel design limits and design conditions of the reactor coolant boundary are not exceeded as a result of anticipated operational occurrences, and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents. The onsite electric power supplies (including the batteries) and the onsite electric distribution system is required to have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure. Electric power from the transmission network to the onsite electric distribution system is required to be supplied by two physically-independent circuits designed and located to minimize the likelihood of their simultaneous failure. Each of these circuits is required to be designed to be available in sufficient time following a loss of all onsite alternating power (ac) power supplies and following the loss of the other offsite electric power circuit, to assure that fuel design limits and design conditions of the RCPB are not exceeded. One of these circuits is required to be available within a few seconds following an accident to assure that core cooling, containment integrity, and other vital safety functions are maintained. In addition, GDC 17 requires provisions to minimize the probability of losing electric power from the remaining electric power supplies as the result of loss of power from the unit, the offsite transmission network, or the onsite power supplies.

GDC 18, “Inspection and Testing of Electric Power System,” requires that electric power systems important to safety be designed to permit appropriate periodic inspection and testing. Criterion XI, “Test Control,” of 10 CFR Part 50, Appendix B, requires a test program be established to ensure that systems important to safety perform satisfactorily. As noted above, 10 CFR 50.36, “Technical Specifications,” requires LCOs be established. This is to assure the availability of required equipment needed to preserve the operability of a minimal set of systems (or safety group) so that (1) fuel design limits and design conditions of the reactor coolant boundary are not exceeded as a result of anticipated operational occurrences, and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

SRs are requirements, included as part of TSs, to assure that the necessary quality of systems and components are maintained and LCOs will be met. When an LCO is not met, either due to component failure or maintenance outage, action is required within a specified time by the TS to restore required equipment to an operable condition. This specified time to take action is referred to as the allowed outage time (AOT) or completion time (CT). The AOT or CT is a temporary relaxation of operability for required equipment, which provides a limited time to fix components and return required equipment to an operable status. Establishing this limited time to fix components is based, primarily, on the reliability of remaining operable required equipment during the short time period of a CT being judged commensurate with reliability when all required equipment is operable.

Regulatory Guide (RG) 1.174, “An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions of Plant-Specific Changes to the Licensing Basis,” and RG 1.177, “An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications,” provide a risk-informed approach for establishing this limited time to fix components. Using a risk-informed approach, the impact from the increased unavailability of required equipment is measured by plant risk via core damage frequency and large early release frequency. When the impact on risk meets the guidelines of RG 1.174 and RG 1.177, the CT can be considered acceptable from a risk perspective. The CT can, therefore, be considered short enough, based on this risk perspective, such that the reliability of remaining operable required equipment can be judged commensurate with the reliability when all required equipment is operable.

Pursuant with 10 CFR 50.36(c), LCOs are specified for required equipment relied upon to preserve a safety group so that (1) fuel design limits and design conditions of the reactor coolant boundary are not exceeded as a result of anticipated operational occurrences, and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents. With few exceptions, these LCOs address single outages of components, trains or subsystems. For any particular system, the LCO does not normally address multiple outages of redundant components, nor does it address the effects of outages of most support systems, such as cooling water, that are relied upon to maintain the operability of the particular system. Multiple outages of components are not explicitly addressed by LCOs because of the large number of combinations of these types of outages that are possible. Instead, general specifications are employed with an explicit definition of the term “operable” to encompass all such cases. These provisions were formulated to assure that no combination of equipment outages can result in the facility operating without the availability of the required minimal set of systems or safety-group needed to accomplish the required safety function.

For this review, the NRC staff will assure that: (1) the TS change will not result in plant operation that will violate any safety limits; (2) the TS change does not affect the operability or testing requirements of the ESFAS; and (3) the licensee has included the testing requirements of the ESFAS elsewhere in the TSs.

3.0 TECHNICAL EVALUATION

3.1 TS 1.0, “Definitions”

The licensee is proposing to revise Technical Specification 1.0, “Definitions,” to correct an editorial error. The Technical Specification definition for “Identified Leakage” is incorrectly numbered as item “1.1.” This definition will be numbered as Item “1.16.” This is an editorial change and has no effect on plant safety. Therefore, the staff considers this acceptable.

3.2 TS 2.2.1, “Limiting Safety Systems Settings - Reactor Trip System Instrumentation Setpoints”

The licensee proposed deleting Functional Unit 17, “Safety Injection Input from ESF,” from TS Table 2.2.1, “Limiting Safety System Settings-Reactor Trip System Instrumentation Setpoints.” The licensee noted that the reason for requesting approval for this TS change was to achieve consistency with a similar change previously reviewed and approved by the NRC.

The NRC staff reviewed the MP3 TSs and found that RTS TS requirements for “Safety Injection Input from ESF” were not included in TS 3.3.1, “Reactor Trip System Instrumentation.” TSs are required to include limiting safety system settings. Section 50.36(c)(1)(ii)(A) of 10 CFR states that “limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions.” Functional Unit 17 ensures that, if a reactor trip has not already been generated by the RTS, the ESF automatic actuation logic will initiate a reactor trip upon any signal that initiates SI. This is a condition of acceptability for loss of coolant accident. The staff concluded that deleting Functional Unit 17 creates a compliance issue with the MP3 TSs. The licensee was made aware of the issue and upon further discussion, the licensee decided, at this time, to withdraw this proposed change to the TSs.

The licensee also proposed changes to TS 2.2.1, to modify the labels for Functional Units 18.b.1 and 18.b.2. These would be modified from “P-10 Input” and “P-13 Input” to “Power Range Neutron Flux, P-10 Input” and “Turbine Impulse Chamber Pressure, P-13 Input,” respectively. These changes are editorial in nature and based on this, the NRC staff finds that the proposed changes do not affect any RTS operability or testing requirements, do not adversely affect plant safety, and meet the requirements of 10 CFR 50.36. Therefore, the staff considers that the proposed TS changes are acceptable.

3.4 TS 3.3.1, Table 4.3-1, “Reactor Trip System Instrumentation Surveillance Requirements”

The licensee proposed to revise the SRs for Functional Unit 6, “Source Range Neutron Flux,” of Table 4.3-1 so that surveillance is required in Modes 3, 4, and 5 only when the RTS breakers are in the closed position and the Control Rod Drive System is capable of rod withdrawal. This revision will make the SRs for Source-Range Neutron Flux consistent with the corresponding operability requirements (Table 3.1-1, Functional Unit 6b). The licensee states:

> The existing surveillance requirement, Table 4.3-1, item 6 requires that the applicable surveillances for the Source Range Neutron Flux Monitor shall be performed at all times during Modes 3, 4, and 5, which is inconsistent with the OPERABILITY requirements for this functional unit. The Source Range Neutron Flux Monitors do not perform a reactor trip function, i.e. they are not required to mitigate an uncontrolled rod withdrawal when the Reactor Trip Breakers Are Open since all rod motion is precluded. The Shutdown Margin Monitors, Technical Specification 3.3.5, “Instrumentation - Shutdown Margin Monitor” are credited with mitigation of the applicable accident, a boron dilution event. The Shutdown Margin Monitors provide the necessary alarm such that operator action can mitigate the consequences of a boron dilution event. Therefore, Table 4.3-1, Item 6 will be revised such that the Source Range Neutron Flux surveillance requirements are applicable in Modes 3, 4, and 5 only when the Reactor Trip System breakers are in the closed position and the Control Rod Drive System is capable of rod withdrawal.

The staff considers the above-described licensee’s proposal acceptable because the requested change would render the SRs consistent with the operability requirements, and the proposed change will continue to comply with the requirements of 10 CFR 50.36.

Also, in regard to other changes to Table 4.3-1, the licensee states:

> Technical Specification 3.3.1, Table 4.3-1, Item 9, “Pressurizer Pressure - Low,” and Item 11, “Pressurizer Water Level - High,” will be revised such that the applicable surveillance requirements are only required to be performed in Mode 1 at power levels above the P-7 interlock (at power) setpoint, consistent with the existing OPERABILITY requirements for the Pressurizer Pressure - Low and Pressurizer Water Level High functional units. The Pressurizer Pressure - Low and Pressurizer Water Level High functional units are blocked by the P-7 interlock below approximately 10 percent of full power to allow reactor startup. Therefore, performance of these surveillance requirements at low power levels below the P-7 interlock setpoint is unnecessary. To support this change a new note, note “*****” will be added to Table 4.3-1 which states, “Above the P-7 (At Power) Setpoint.”

The staff concurs with the licensee's proposal and considers the proposed SRs for low pressurizer pressure acceptable because they are consistent with both the associated existing MP3 operability requirements (Table 3.1-1, Functional Unit 9) and the operability requirements of the Standard Technical Specifications (STS). The proposed SRs for high pressurizer water level, while not consistent with the SRs of the STS (the threshold requirement is set at high power rather than at low power), are consistent with the associated existing MP3 operability requirements (Table 3.1-1, Functional Unit 11). The existing operability requirements require operability at high power, and not at low power. Based on these reasons the staff approves the change because the change will continue to comply with the requirements of 10 CFR 50.36.

3.4 TS 3.3.2, "Engineered Safety Features Actuation System Instrumentation"

The licensee contends that the proposed TS 3.3.2 change specifies more appropriate actions if multiple channels of 4 kV Bus Undervoltage-Grid degraded Voltage or Undervoltage-Loss of Voltage instrumentation are inoperable.

Current TS 3.3.2

ACTION 20 - With the number of OPERABLE channels one less than the Total number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.2.1.

If more than one channel is inoperable, current TS require plant shutdown.

When one of four channels is inoperable, current ACTION 20a allows six hours to place the inoperable channel in its tripped condition. Plant operation is, thus, allowed for six hours in either a one-of-three or a two-of-three channels for actuation logic (depending on how the logic channel failed) and, after six hours, in a one-of-three channels for actuation logic configuration. And, for four hours, ACTION 20b allows the inoperable channel to be changed from its tripped condition to its bypassed condition. Plant operation is, thus, allowed for four hours in a two-of-three channels for actuation logic to permit surveillance testing of other channels. While testing, the logic may be in a one-of-two channels for actuation configuration.

Proposed TS 3.3.2

ACTION 27a. With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

1. The inoperable channel is placed in the tripped condition within 6 hours, and

2. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 27b. With the number of OPERABLE channels one less than the Minimum Channels required OPERABLE:

1. Place one channel in bypass and other channel in trip condition within one hour and restore one channel to OPERABLE status within 48 hours,

OR

2. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

If more than two channels are inoperable, the proposed TS requires plant shutdown.

When one channel is inoperable, ACTION 27a is the same as current ACTION 20; thus, no change has been proposed. ACTION 27b, however, changes ACTION 20 to allow a more appropriate action (as opposed to plant shutdown) when two channels are inoperable. ACTION 27b allows one hour to place one channel in bypass and the other channel in trip and 48 hours to restore to operable status one of the two inoperable channels. Plant operation would, thus, be allowed for one hour in either a zero-of-two or one-of-two channels for actuation logic; 48 hours in a one-of-two channels for actuation logic; or in a one-of-three channels for actuation logic configuration.

Operation in a one-of-two or one-of-three configuration assures the accomplishment of the safety function (i.e., trip or isolation of offsite and transfer to onsite). A one-of-two or one-of-three configuration meets regulatory requirements of GDC 17 and are considered acceptable. Plant operation for one hour with a potential zero-of-two logic for trip conveys that for one hour there will be some greater potential for spurious actuation. Spurious actuation, if it has (or were) to occur, does not result in a reactor trip or transient. The TS allowable time of one hour is considered acceptable because (given that there has not been an actuation due to the potential zero-of-two logic) the logic remains in a one-of-two logic which meets regulatory requirements of GDC 17.

3.5 <u>TS 3.8.3.1, "Onsite Power Distribution - Operating"</u>

The licensee contends that TS 3.8.3.1 as currently written has not clearly identified the OPERABILITY requirements for the applicable emergency buses. The proposed changes will ensure that the inoperable electrical bus is restored to OPERABLE status prior to exiting the applicable TS Action statement(s). (In the following discussion on the changes to TS 3.8.3.1, words in **bold** indicate the specific changes being addressed.)

Current TS 3.8.3.1

3.8.3.1 The identified electrical busses shall be **energized** in the specified manner:

ACTION:

a. With one of the required trains of ac emergency buses **not fully energized, reenergize the division** within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS:

4.8.3.1 The specified busses shall be determined **energized in the required manner** at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

Proposed TS 3.8.3.1

3.8.3.1 The identified electrical busses shall be **OPERABLE** in the specified manner:

ACTION:

a. With one of the required trains of ac emergency busses not **OPERABLE, restore the inoperable train to OPERABLE status** within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS:

4.8.3.1 The specified busses shall be determined **OPERABLE in the specified manner** at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

The licensee proposes to change the terminology used in TS 3.8.3.1 for distribution buses from "energized in the specified manner" to "OPERABLE in the specified manner." The licensee contends that the OPERABILITY requirements for the applicable emergency busses are not clearly identified. Also, terminology utilized as part of associated actions and SRs are proposed for modification in order to maintain consistency of terminology within TS 3.8.3.1.

The proposed change conveys that an inoperable electrical bus is restored to OPERABLE status prior to exiting the applicable TS Action statement(s). The proposed change does not revise any existing OPERABILITY requirements for the applicable electrical busses, nor does the proposed change affect any testing requirements.

The staff agrees that the proposed change clarifies operability requirements for applicable emergency busses and does not revise any existing operability requirements for electrical distribution systems. The staff, therefore, concludes that the new TS 3.8.3.1, with the proposed changes included, is acceptable.

3.6 TS 3.7.7, "Control Room Emergency Ventilation System"

Currently, TS 3.7.7, "Control Room Emergency Ventilation System," Surveillance Section 4.7.7 e. 2) states:

> Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch Water Gauge at less than or equal to a pressurization flow of 230 cfm relative to adjacent areas and outside atmosphere during positive pressure system operation; and

After the change, TS 3.7.7, "Control Room Emergency Ventilation System," Surveillance Section 4.7.7 e. 2) will state:

> Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch Water Gauge at less than or equal to a pressurization flow of 230 cfm relative to adjacent areas and outside atmosphere during the filtered pressurization mode of operation; and

The phrase "positive pressure system" in the current TS is being replaced by the phrase "the filtered pressurization mode of" in the proposed change of the TS. The licensee decided to use the latter phrase in order to be consistent throughout the TSs with the appropriate characterization of the process. The staff considers this to be an editorial change and that the change is acceptable. There is no impact on system operation or public health and safety due to this change.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Connecticut State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, and changes SRs. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (67 FR 63692). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Letter from J. Alan Price, Site Vice President, Millstone to USNRC, "Millstone Nuclear Power Station, Unit No. 3, License Basis Document Change Request 3-5-02, Limiting Safety System Settings and Instrumentation," August 7, 2002.

2. Letter from Vernon L. Rooney, USNRC to Ted C. Feigenbaum, Northeast Utilities Service Company, "Amendment No. 129 to Facility Operating License NPF-49," June 27, 1996.

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Date: September 14, 2004