September 16, 2002

Mr. J. A. Price
Site Vice President - Millstone
Dominion Nuclear Connecticut, Inc.
c/o Mr. David A. Smith
Rope Ferry Road
Waterford, CT 06385

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 3 - ISSUANCE OF AMENDMENT RE: REVISED FINAL SAFETY ANALYSIS REPORT LICENSING BASIS FOR THE POST-ACCIDENT OPERATION OF THE SUPPLEMENTARY LEAKAGE COLLECTION AND RELEASE SYSTEM (TAC NO. MB3700)

Dear Mr. Price:

The Commission has issued the enclosed Amendment No. 211 to Facility Operating License No. NPF-49 for the Millstone Power Station, Unit No. 3 (MP3), in response to your application dated June 6, 1998, as supplemented by letters dated April 5, 1999; April 7, April 19, July 31, and September 28, 2000; March 19, June 11, September 21, and December 20, 2001.

The amendment changes the licensing basis for the post-accident operation of the Supplementary Leakage Collection and Release System as described in the MP3 Final Safety Analysis Report.

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, Sr. 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. 211 to NPF-49
2. Safety Evaluation

cc w/encls: See next page

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Accession Number: ML022470399 **/C/ via e-mail
*SE provided 9/7/99 and 5/10/02, no major changes made.

OFFICE	PDI-2/PM	PDI-2/LA**	IQMB/BC*	SPSB/SC(A)*	OGC	PDI-2/SC(A)
NAME	VNerses	TLClark	RGallo	MReinhart	CBray	JAndersen
DATE	9/5/02	09/05/02	9/7/99	5/10/02	9/9/02	9/16/02

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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. 211
License No. NPF-49

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

   A. The application for amendment by the applicant dated June 6,1998, as supplemented by letters dated April 5, 1999; April 7, April 19, July 31, and September 28, 2000; March 19, June 11, September 21, and December 20, 2001, 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, changes to the Final Safety Analysis Report (FSAR) to reflect a revised licensing basis for post-accident operation of the Supplementary Leakage Collection and Release System are authorized.

3. This license amendment is effective as of the date of issuance, and shall be implemented within 60 days of issuance. Implementation includes revision of the FSAR as described in the licensee's incoming application and the staff's safety evaluation.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

James W. Andersen, Acting Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: September 16, 2002

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 211

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 June 6, 1998, as supplemented by letters dated April 5, 1999; April 7, April 19, July 31, and September 28, 2000; March 19, June 11, September 21, and December 20, 2001, Northeast Nuclear Energy Company (NNECO the then licensee) and Dominion Nuclear Connecticut, Inc. (licensee or DNC) submitted a request to make a change in the licensing basis for post-accident operation of the Supplementary Leakage Collection and Release System (SLCRS) as described in the Final Safety Analysis Report (FSAR) of the Millstone Power Station, Unit No. 3 (MP3). The April 5, 1999; April 7, April 19, July 31, and September 28, 2000; and March 19, June 11, September 21, and December 20, 2001, letters provided clarifying information that was within the scope of the original application and did not change the staff's proposed no significant hazards consideration determination.

At the time of the application, NNECO was the licensed operator of MP3. On March 31, 2001, the majority of the owners of MP3 transferred their ownership interests in MP3 to DNC. By letter dated April 2, 2001, DNC requested that the U.S. Nuclear Regulatory Commission (NRC) continue to review and act upon all requests before the NRC that had been submitted by NNECO.

2.0 BACKGROUND

The June 6, 1998, submittal requested changes to the licensing basis for the post-accident operation of the SLCRS as described in the MP3 FSAR. The SLCRS is used to maintain a negative pressure relative to atmospheric pressure in the secondary containment by collecting air from the enclosure building and connecting areas, filtering it to remove iodine, and discharging to the atmosphere. The licensee has identified potential release pathways from secondary containment to the environment that could bypass the SLCRS filter following a design-basis accident due to non-nuclear safety grade (NNS) exhaust fan operation after the accident. These additional pathways are not included in the current design-basis accident dose analyses as documented in the MP3 FSAR, therefore making them non-conservative.

By letter dated April 19, 2000, the licensee requested that NRC temporarily suspend review of the license amendment request. This suspension was prompted by the licensee's internally identified concern related to the adequacy of the original licensing basis documentation supporting the determination of spray coverage within the free volume of the containment. The licensee's proposed resolution of this concern and subsequent impacts to the radiological analyses were submitted to the NRC by letter dated September 28, 2000.

During an August 29, 2001, conference call between the NRC staff and DNC personnel, the staff raised further questions related to the analysis of containment spray mixing rate during a design-basis loss-of-coolant accident (LOCA). Specifically, the analysis uses an enhanced mixing model to quantify the spray coverage and the time-dependent mixing rate between the unsprayed regions and the sprayed regions. During the conference call, the staff indicated that based on their review, additional information would be needed for approval of the application of the mixing model relied upon in the MP3 license amendment request. Containment spray coverage and the derived mixing rates are used to establish the iodine removal efficiency of the sprays and, therefore, have a direct impact on the associated post-accident dose assessment.

Subsequently, in a letter dated September 21, 2001, DNC committed to submit a revised dose assessment using an NRC-approved methodology and requested that the staff suspend their review until a revised dose assessment related to the dose consequences was completed and formally submitted to the NRC. The licensee's letter dated December 20, 2001, provided the information so the staff could complete the review that had been suspended.

3.0 EVALUATION

Two FSAR Chapter 15 design-basis accident (DBA) radiological consequences analyses were affected by the change to the SLCRS design-basis, the LOCA and the Rod Ejection Accident. The following discusses the changes made by the licensee and the staff's evaluation.

3.1 Relative Concentration (X/Q) Estimates

The licensee has provided X/Q estimates for postulated design-basis accident releases to the exclusion area boundary (EAB), low population zone (LPZ), and control room air intake. X/Q values calculated for the EAB and LPZ use the methodology described in Regulatory Guide (RG) 1.145 (Ref. 1) and adapted from RG 1.3 (Ref. 2) for fumigation conditions. X/Q values for ground level releases to the control room air intake were estimated using the Murphy-Campe (Ref. 3) methodology. Estimates for the MP3 elevated release from the MP1 stack were made using the RG 1.145 methodology assuming both fumigation and non-fumigation conditions.

Some of the X/Q values provided update historical values using computer assessment to replace the original hand calculations. The licensee had observed some inconsistencies in the statistical analyses between the older and newer calculations and determined that the X/Q values should be based on the modern techniques. In addition, new control room X/Q values were calculated for several other postulated release locations that were recently identified.

The postulated release from the Main Steam Valve Building (MSVB) does not meet the diffuse source option criteria of the Murphy-Campe methodology since the difference in elevation between the release and receptor height is less than 30 percent of the building height. However, flow that would carry effluent to the control room intake would meet several aerodynamic obstructions, resulting in enhanced mixing. DNC staff also made limited confirmatory approximations for the MSVB release point using the ARCON96 (Ref. 4) methodology.

Based on the licensee's use of accepted methodologies and a qualitative review of the proposed values, the NRC staff concludes that the proposed X/Q values are acceptable for use in the dose assessment associated with this amendment request. These values are listed in Table 1.

3.2 Loss-of-Coolant-Accident (LOCA)

3.2.1 Operator Actions

The NRC staff's review focused on verifying that the licensee had considered aspects of human performance which could influence the operators ability to perform the proposed actions to secure the five fans within the prescribed timeframe allotted following a postulated LOCA. The staff used the information regarding operator actions described in Generic Letter (GL) 91-18, Revision 1, "Resolution of Degraded and Nonconforming Conditions and on Operability," 1997, ANSI/ANS 58.8, "Time Response Design Criteria for Safety Related Operator Actions," 1984 (ANSI-58.8), and Information Notice 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modification of Operator Actions, Including Response Times," 1997.

Based on these guidelines, the staff's review of the licensee's proposed amendment request included: (1) the specific operator actions required; (2) the potentially harsh or inhospitable environmental conditions expected; (3) a general discussion of the ingress/egress paths taken by the operators to accomplish functions; (4) the procedural guidance for required actions; (5) the specific operator training necessary to carry out actions, including any operator qualifications required to carry out actions; (6) any additional support personnel and/or equipment required by the operator to carry out actions; (7) a description of information required by the control room staff to determine whether such operator action is required, including qualified instrumentation[1] used to diagnose the situation and to verify that the required action has successfully been taken; (8) the ability to recover from credible errors in performance of manual actions, and the expected time required to make such a recovery; and (9) consideration of the risk significance of the proposed operator actions.

Proposed revision to FSAR Section 6.4.3, states in part that, in the event of a LOCA, operators are credited with securing selected NNS fans within 20 minutes following at least 1 hour of operation in the pressurization mode. After 1 hour, control room ventilation can be realigned from pressurization to recirculation mode. The licensee performed an operator activity timing study to verify the time to completion of the task of securing the NNS fans. For the study, the licensee assumed only one operator would be available to perform all the activities associated with securing the fans. The operator would, per procedure, first be directed to open the

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[1]In accordance with RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident," Revision 3, qualification of the instrumentation relied upon by the operators may be an important review issue. RG 1.97 defines Type A variables as "those variables to be monitored that provide the primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their functions for design-basis accident events."

breakers for the five NNS fans before proceeding to the Control Building upper level to open the air inlet valves (3HCV*A0V25&26) as part of the realignment task. The results of the time study indicated that the total elapsed time to manually open the breakers for the five NSS fans was approximately 15 minutes, including the time required for the control room operator to initiate the appropriate operating procedure and direct the auxiliary operator to perform the actions, and transit time of the auxiliary operator between the control room and the breaker location and return to the control room.

As part of the review, the staff requested that the licensee describe the indications and controls (including location) which are relied upon by the operators to (1) determine that the fans must be isolated, and (2) to accomplish the required actions. For any actions taken outside of the control room, the licensee was requested to describe the presumed environmental conditions to which an operator would be subjected (heat, humidity, lighting, radiological, etc.) and describe how potential barriers to successful operator performance were considered and dispositioned.

The licensee provided information which indicated that by procedure the NSS fans are de-energized regardless of their current status at the time of the initiation of the transient. As a result, no fan controls or operating status indications within the control room are required to determine if the actions to open the NNS fan breakers are necessary. The licensee indicated that the only information required by the operations crew to perform these actions is a timer within the control room to ensure that the NNS fans are secured within the 1-hour and 20-minute timeframe following the LOCA initiation. The local auxiliary operator actions required to manually open the NNS fan breakers require the auxiliary operator to access the breaker cubicles located within several motor control centers (MCC) and load centers within the Service Building and Auxiliary Building, respectively. Each of the respective MCC/load centers are identified and each of the breaker cubicles are numbered to facilitate operator action.

The licensee evaluated the environmental conditions which the auxiliary operator might be subjected to during the manual actions of securing the NNS fans. The licensee determined that the operator would not be adversely affected by the postulated environmental conditions based on a qualitative assessment which considered the equipment breaker locations and times required to perform the specified actions at those locations. Potential radiological conditions were evaluated in accordance with NUREG-0737 guidance. Based on the planned ingress/egress routes and time estimates for the auxiliary operator to perform the actions to secure the NNS fans, the maximum dose calculated by the licensee was within the NUREG-0737 guidelines. Additionally, the licensee indicated that emergency lighting would be available for ingress/egress should the normal lighting fail to operate during the LOCA recovery.

The licensee described procedural modifications which were implemented to direct the operators to secure the NNS fans, which included adding a step to the Control Building Heating, Ventilation, and Air Conditioning operating procedure to have the operators refer to an attachment which listed the specific NNS fans that needed to be de-energized following a LOCA event. The staff verified that the licensee had considered its operator training programs and verified that its training was sufficient to ensure that those actions specified in the procedures could be accomplished by the operating crews.

Finally, the licensee performed an analysis to determine what other breakers or controls are in close proximity (i.e., adjacent to, or within the normal reach of the operator) to the NNS fan breakers and what, if any, effects would result on mitigation if these other breakers or controls were inadvertently manipulated. The licensee's analysis indicated that inadvertent manipulation of the adjacent equipment within the NNS fan breaker cubicles would have no effect on the successful mitigation of the LOCA event.

In summary, the licensee has provided the staff with sufficient information to demonstrate that the operators are capable of performing the required functions to secure selected NNS-grade fans within a prescribed timeframe following a LOCA and under conditions presumed to exist at the time the actions are required. The licensee has developed and implemented adequate procedures and training for the conduct of such actions. The proposed use of operator actions, as described in the proposed license amendment and revision to the licensee's FSAR are, therefore, acceptable.

3.2.2 Technical Evaluation

The licensee has performed three separate leakage scenarios, each one limiting for the dose calculation offsite, in the control room, and in the technical support center (TSC). For calculation of the offsite dose, the licensee assumed the NNS fans continue operating, with the associated leakage through boundary dampers, for the entire 30-day dose analysis period. For the control room habitability analysis, the licensee assumed the NNS fans continue to operate for 1 hour and 20 minutes, at which time the fan breaker is assumed to have been manually tripped. At 1 hour and 40 minutes, the control room ventilation system is realigned to the filtered recirculation mode and the control room is repressurized. The TSC habitability analysis assumes that the NNS fans continue to operate until manually secured by an operator 1 hour and 20 minutes after the accident. This proposed use of operator actions was found acceptable as described in detail in Section 3.2.1. Therefore, the staff found the licensee's radiological analysis assumptions that take credit for these operator actions acceptable. For example, the duration and rate of the release through the SLCRS bypass pathways are based on operator actions to secure certain NNS fans. It is also noted that all scenarios assume offsite power is available for the duration of the accident.

The licensee made some additional changes to the LOCA dose analysis. The iodine species composition and iodine core inventory available for release from the containment have been changed to be those assumed in U.S. Atomic Energy Commission (USAEC) Technical Information Document (TID) 14844 (Ref. 5), and are, therefore, acceptable. In addition, the licensee used dose conversion factors from the International Commission on Radiation Protection Publication 30 (ICRP-30), which the staff also finds acceptable.

The iodine removal coefficients (lambdas) in Insert G of the licensee's submittal dated April 19, 2000, were verified by the staff using the information provided by the licensee. All the coefficients were calculated using the methodology described in Standard Review Plan (SRP) Section 6.5.2, "Containment Spray as a Fission Product Cleanup System." NRC staff review has verified that the licensee's calculated values for these coefficients are acceptable.

The licensee's analysis submitted in 1998 used a Stone and Webster Engineering Corporation proprietary methodology to determine calculated input values for containment mixing due to sprays. The staff had many discussions with the licensee and its contractor about this methodology. In particular, during a conference call on August 29, 2001, the staff requested more information about this methodology, because the docketed information did not support a finding of acceptability. By letter dated December 20, 2001, the licensee supplemented the original submittal with a revised dose analysis that uses the NRC-accepted SRP 6.5.2 mixing rate assumption of two unsprayed region volumes per hour between the sprayed and unsprayed regions of the containment. This revised analysis replaced the previously submitted one using variable mixing rates determined from the proprietary methodology. As a result of this change, the proprietary mixing rate methodology will no longer be discussed here.

The staff determined that the licensee's analysis and assumptions followed guidance given in RG 1.4 (Ref. 6) and SRP 15.6.5 (Ref. 7) for LOCA radiological analyses and, therefore, are acceptable. Using the licensee's assumptions, the staff performed independent calculations confirming the licensee's doses for the LOCA. The licensee's dose results do not exceed the dose limits in 10 CFR Part 100 for offsite doses, and do not exceed the dose limits given in 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 19 for control room doses. The staff has determined that the TSC habitability analysis assumptions are acceptable, and the dose results are also within GDC-19 limits. The licensee's dose results and regulatory acceptance criteria are given in Table 2.

3.3 Control Room Habitability

The licensee's analysis for control room habitability is evaluated for the limiting LOCA, and the results were previously discussed. The licensee used SRP Section 6.4, "Control Room Habitability System" to adjust the unfiltered inleakage value that was previously used, which was based on the filtered intake flow rate. This assumed unfiltered inleakage rate is not based on testing of the control room envelope. Because the radioactivity release through the SLCRS bypass pathway is considered to be minimal, as can be seen by the resulting change in offsite dose, the staff did not pursue the matter further. The staff considers the licensee's assumptions for control room unfiltered inleakage to be acceptable for use in the current radiological dose consequences analysis for this license amendment request.

The staff is currently developing regulatory guidance regarding control room habitability, including surveillance testing of unfiltered inleakage. In addition, the Nuclear Energy Institute (NEI) has developed an industry initiative document on control room habitability NEI 99-03, "Control Room Habitability Assessment Guidance." The staff's acceptance of this MP3 control room unfiltered inleakage assumption does not foreclose on any future generic regulatory actions that may become applicable to MP3 in this regard.

3.4 Rod Ejection Accident (REA)

The licensee also incorporated the additional SLCRS bypass leakage into the design-basis rod ejection accident radiological dose consequences analysis. The revised REA analysis also uses the dose conversion factors from ICRP-30. The staff has determined the licensee's assumptions for the REA radiological consequences analysis followed guidance given in RG 1.77 (Ref. 8) and SRP 15.4.8 (Ref. 9) and are, therefore, acceptable. The staff performed independent calculations to confirm the licensee's dose results for the REA. The licensee's offsite dose results meet the acceptance criteria given in SRP 15.4.8, Appendix A, being well within (25% of) the dose limits in 10 CFR Part 100. The control room doses due to the design-basis LOCA remain bounding. The licensee's dose results and regulatory acceptance criteria are also given in Table 2.

3.5 Summary

Based on the licensee's use of acceptable methodologies and assumptions as previously discussed and the staff's confirmation of the licensee's dose results, the staff has determined that the licensee's revised design-basis accident radiological consequences analyses for the LOCA and rod ejection accident, which take into account additional SLCRS bypass release pathways, are acceptable. The licensee has demonstrated that the calculated dose consequences of a postulated design-basis LOCA are within 10 CFR Part 100 dose limits for offsite doses and 10 CFR Part 50, Appendix A, GDC-19 dose limits with regard to control room habitability. The offsite dose consequences of a postulated rod ejection accident are demonstrated to be well within (25% of) the dose limits in 10 CFR Part 100, with the LOCA control room doses bounding. Therefore, the staff finds the revision to the MP3 licensing and design-basis to be acceptable with regard to the radiological consequences of design-basis accidents.

Table 1

MP3 RELATIVE CONCENTRATION (X/Q) VALUES

Offsite X/Q Values

a. EAB X/Q values

Ground level release - Containment 0 - 2 hr.	5.42 E-4* s/m^3
Ground level release - Ventilation vent 0 - 2 hr.	4.3 E-4 s/m^3
Elevated release - Millstone Stack 0 - 2 hr.	1.0 E-4 s/m^3

b. LPZ X/Q values

Ground level release - Containment 0 - 8 hr.	2.91 E-5 s/m^3
Ground level release - Ventilation vent	
0 - 8 hr.	2.91 E-5 s/m^3
8 - 24 hr.	1.99 E-5 s/m^3
1 - 4 days	8.66 E-6 s/m^3
4 - 30 days	2.63 E-6 s/m^3
Elevated release - Millstone stack	
0 - 4 hr.	2.69 E-5 s/m^3
4 - 8 hr.	1.07 E-5 s/m^3
8 - 24 hr	6.72 E-6 s/m^3
1 - 4 days	2.46 E-6 s/m^3
4 - 30 days	5.83 E-7 s/m^3

Unit 3 Control Room X/Q Values

1. Ground level release - Containment	
0 - 8 hr	1.52 E-3 s/m^3
8 - 24 hr.	8.53 E-4 s/m^3
1 - 4 days	2.59 E-4 s/m^3
4 - 30 days	3.21 E-5 s/m^3
2. Elevated release - Millstone Stack	
0 - 4 hr.	1.39 E-4 s/m^3
4 - 8 hr.	3.23 E-5 s/m^3
8 - 24 hr.	1.56 E-5 s/m^3
1 - 4 days	1.92 E-6 s/m^3
4 - 30 days	1.32 E-7 s/m^3
3. Ground level release - Ventilation Vent	
0 - 8 hr.	3.75 E-3 s/m^3
8 - 24 hr.	2.28 E-3 s/m^3
1 - 4 days	7.43 E-4 s/m^3
4 - 30 days	9.69 E-5 s/m^3
4. Unit 3 Main Steam Valve Building	
0 - 8 hr.	5.78 E-3 s/m^3
8 - 24 hr.	3.20 E-3 s/m^3
1 - 4 days	9.52 E-4 s/m^3
4 - 30 days	9.16 E-5 s/m^3
5. Unit 3 Engineered Safety Features Building	
0 - 8 hr.	4.86 E-3 s/m^3
8 - 24 hr.	2.69 E-3 s/m^3
1 - 4 days	8.00 E-4 s/m^3
4 - 30 days	6.77 E-5 s/m^3
6. Unit 3 Refueling Water Storage Tank	
0 - 8 hr.	N/A
8 - 24 hr.	8.53 E-4 s/m^3
1 - 4 days	4.32 E-4 s/m^3
4 - 30 days	8.03 E-5 s/m^3

***** 5.42 E-4 = 5.42 x 10^{-4}

Table 2

LICENSEE-CALCULATED MP3 DESIGN-BASIS ACCIDENT DOSES INCLUDING SLCRS BYPASS LEAKAGE PATHWAYS

Offsite Dose Consequences

	EAB (rem)	LPZ (rem)	Acceptance Criterion (rem)
LOCA			
Thyroid	80	27	300
Whole Body	4.0	0.83	25
Rod Ejection Accident			
Thyroid	10	10.3	75
Whole Body	0.1	0.04	6

MP3 Control Room and TSC Habitability

	MP-3 CR(rem)	TSC (rem)	Acceptance Criterion (rem)
LOCA			
Thyroid	26	4.4	30
Whole Body	1.1	0.67	5

4.0 <u>STATE CONSULTATION</u>

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 <u>ENVIRONMENTAL CONSIDERATION</u>

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact has been prepared and was published in the *Federal Register* on July 23, 2002 (67 FR 48211). Accordingly, based upon the environmental assessment, the staff has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

6.0 <u>CONCLUSION</u>

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 <u>REFERENCES</u>

1. USNRC, Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*, November 1982

2. USNRC, Regulatory Guide 1.3*, Assumptions Used for Evaluating the Potential Radiological Consequences of Loss of Coolant Accident for Boiling Water Reactors*, June 1974

3. Murphy, K.G. and Campe, K.W., *Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19*, published in proceedings of 13th AEC Air Cleaning Conference

4. Ramsdell, J.V., Jr. and Simonen, C.A., 1997, *Atmospheric Relative Concentrations in Building Wakes*, NUREG/CR-6331, Rev.1, U.S. Nuclear Regulatory Commission, Washington, DC

5. TID-14844, *Calculation of Distance Factors for Power and Test Reactor Sites,* 1962, U.S. Atomic Energy Commission, Washington, DC

6. USNRC, Regulatory Guide 1.4*, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors*, June 1974

7. USNRC, NUREG-0800, Standard Review Plan, Section 15.6.5, *Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary*, July 1981

8. USNRC, Regulatory Guide 1.77, *Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors*, May 1974

9. USNRC, NUREG-0800, Standard Review Plan, Section 15.4.8, Appendix A, *Radiological Consequences of a Control Rod Ejection Accident (PWR)*, July 1981

Principal Contibutors: M. Hart
L. Brown
G. Galletti
V. Nerses

Date: September 16, 2002

cc:

Ms. L. M. Cuoco
Senior Nuclear Counsel
Dominion Nuclear Connecticut, Inc.
Rope Ferry Road
Waterford, CT 06385

Edward L. Wilds, Jr., Ph.D.
Director, Division of Radiation
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

First Selectmen
Town of Waterford
15 Rope Ferry Road
Waterford, CT 06385

Mr. P. J. Parulis
Manager - Nuclear Oversight
Dominion Nuclear Connecticut, Inc.
Rope Ferry Road
Waterford, CT 06385

Mr. W. R. Matthews
Vice President and Senior
Nuclear Executive - Millstone
Dominion Nuclear Connecticut, Inc.
Rope Ferry Road
Waterford, CT 06385

Ernest C. Hadley, Esquire
P.O. Box 1104
West Falmouth, MA 02574-1104

Mr. John Markowicz
Co-Chair
Nuclear Energy Advisory Council
9 Susan Terrace
Waterford, CT 06385

Mr. Evan W. Woollacott
Co-Chair
Nuclear Energy Advisory Council
128 Terry's Plain Road
Simsbury, CT 06070

Mr. D. A. Christian
Senior Vice President - Nuclear Operations
and Chief Nuclear Officer
Innsbrook Technical Center - 2SW
5000 Dominion Boulevard
Waterford, CT 06385

Senior Resident Inspector
Millstone Power Station
c/o U.S. Nuclear Regulatory Commission
P. O. Box 513
Niantic, CT 06357

Mr. G. D. Hicks
Director - Nuclear Station Safety
and Licensing
Dominion Nuclear Connecticut, Inc.
Rope Ferry Road
Waterford, CT 06385

Mr. D. A. Smith
Manager - Licensing
Dominion Nuclear Connecticut, Inc.
Rope Ferry Road
Waterford, CT 06385

Ms. Nancy Burton
147 Cross Highway
Redding Ridge, CT 00870

cc:

Mr. William D. Meinert
Nuclear Engineer
Massachusetts Municipal Wholesale
  Electric Company
Moody Street
P.O. Box 426
Ludlow, MA 01056

Mr. S. E. Scace
Director - Nuclear Engineering
Dominion Nuclear Connecticut, Inc.
Rope Ferry Road
Waterford, CT 06385

Mr. M. J. Wilson
Manager - Nuclear Training
Dominion Nuclear Connecticut, Inc.
Rope Ferry Road
Waterford, CT 06385