April 29, 2002

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

SUBJECT: MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: STEAM GENERATOR TUBE RUPTURE RADIOLOGICAL DOSE CONSEQUENCES (TAC NO. MB0866)

Dear Mr. Price:

The Commission has issued the enclosed Amendment No. 265 to Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Unit No. 2 (MP2), in response to your application dated December 21, 2000, as supplemented June 29, 2001.

The amendment revises the MP2 Final Safety Analysis Report, Chapter 14, description of the Steam Generator Tube Rupture event and the associated radiological dose consequences.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be published in the Federal Register.

Sincerely,

/RA/

Richard B. Ennis, Sr. Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures:
1. Amendment No. 265 to DPR-65
2. Safety Evaluation
3. Notice of Issuance

cc w/encls: See next page

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DOMINION NUCLEAR CONNECTICUT, INC.

DOCKET NO. 50-336

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 265
License No. DPR-65

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

   A. The application for amendment by the applicant dated December 21, 2000, as supplemented June 29, 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) Chapter 14 to reflect a revised description of the Steam Generator Tube Rupture event and the associated radiological dose consequences, as set forth in the licensee's application dated December 21, 2000, and supplemented June 29, 2001, are authorized.

3. This license amendment is effective as of the date of issuance, and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Date of Issuance: April 29, 2002

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 265

TO FACILITY OPERATING LICENSE NO. DPR-65

DOMINION NUCLEAR CONNECTICUT, INC.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

1.0 INTRODUCTION

By letter dated December 21, 2000, as supplemented June 29, 2001, Dominion Nuclear Connecticut, Inc., (Dominion or the licensee), submitted a license amendment request as the result of changes to the assumptions and methods used in the updated thermal hydraulic analysis and in the updated dose consequence analysis of the Steam Generator Tube Rupture (SGTR) event. The requested changes would revise the Millstone Nuclear Power Station, Unit No. 2 (MP2) Final Safety Analysis Report (FSAR), Chapter 14, description of the SGTR event and its associated radiological dose consequences. The changes are not the result of hardware changes to the plant or changes in operating practices. Rather, the changes are the result of incorporating a postulated loss of offsite power (LOOP) into the event analyses as well as revised assumptions and analysis methodology. The proposed FSAR changes show that the postulated dose consequences for the updated SGTR analysis are higher than the dose consequences for the previous analysis which requires, under the provisions of 10 CFR 50.59, that the Nuclear Regulatory Commission (NRC) staff approve the revised analysis before the licensee can implement the proposed change. The June 29, 2001, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

On December 30, 1983, the NRC issued Amendment No. 90 to the MP2 license that included the Technical Specification changes required for cycle 6 operation. The Safety Evaluation related to the amendment stated that the licensee did not evaluate an SGTR concurrent with a LOOP as required by General Design Criterion (GDC) 17, "Electric Power Systems," of Appendix A to 10 CFR Part 50,"General Design Criteria for Nuclear Power Plants." Therefore, approval of Amendment No. 90 was based, in part, on the licensee providing confirmatory analysis to demonstrate that an SGTR concurrent with a LOOP would not substantially alter the conclusions for an SGTR without a LOOP and that continued operation of the plant would not endanger the health and safety of the public.

Northeast Nuclear Energy Company (NNECO), the then licensee for MP2, submitted additional information to the NRC dated September 14, 1984, and January 2 and November 8, 1985, that included the results of the confirmatory analysis for an SGTR with a LOOP. The additional information was provided to address the staff's concerns in the Safety Evaluation for Amendment No. 90. As discussed in the licensee's submittal dated December 21, 2000, the additional information was provided to the NRC for information only to show that the SGTR analysis with a LOOP demonstrates acceptable results. As such, the MP2 licensing basis of the SGTR analysis was not changed to include a LOOP at that time. In a letter dated January 21, 1987, the NRC staff accepted the results of the licensee's confirmatory analysis for a SGTR with a LOOP and considered the issue to be closed.

In a license amendment request dated November 13, 1998, as supplemented on September 16, 1999, NNECO informed the NRC that the SGTR analysis had been updated to resolve some discrepancies which were identified as part of the MP2 Configuration Management Program. The letter stated that the associated proposed FSAR changes showed that the radiological dose consequences for the updated SGTR analysis were higher than the dose consequences for the previous analysis. Therefore, the results of the updated analysis represented an increase in the consequences of a previously analyzed accident and were deemed to involve an unreviewed safety question (USQ) which, under the provisions of 10 CFR 50.59, required prior NRC review and approval before the proposed FSAR changes could become part of the revised licensing basis.

Subsequent discussions between NNECO and the NRC concluded that there was a need to update the SGTR analysis to include a LOOP (i.e., consistent with GDC-17) and include the results of this analysis as part of the MP2 licensing basis. Therefore, by letter dated January 25, 2000, the licensee withdrew the license amendment request dated November 13, 1998, and informed the NRC that a new license amendment request would be submitted that would incorporate a LOOP into the SGTR analysis. The licensee's new license amendment request, submitted by letter dated December 21, 2000, as supplemented June 29, 2001, provided the results of the revised SGTR analysis including a LOOP along with the proposed FSAR changes.

As described by the licensee's application, an SGTR accident is a failure of the barrier between the Reactor Coolant System (RCS) and the Main Steam System. The integrity of this barrier is significant from the standpoint of radiological safety in that a leaking steam generator tube allows the transfer of reactor coolant into the main steam system. Radioactivity contained in the reactor coolant mixes with water in the shell side of the affected steam generator. This radioactivity is transported via the main steam dump and bypass system to the condenser or directly to the atmosphere via the atmospheric dump and safety valves.

The SGTR accident is more thoroughly described in Section 14.6.3 of the FSAR. The assumptions used in the current analysis are given in Section 14.6.3.5 of the FSAR and the analysis parameters are given in Table 14.6.3-5 of the FSAR. The estimated dose consequences of the current SGTR analysis are given in Table 14.6.3-6 of the FSAR and in the June 29, 2001, letter which provided responses to the staff's request for additional information.

According to the licensee's current SGTR analysis with offsite power available, the radioactive contents are released primarily via the condenser air ejector that is aligned to the Unit No.1 stack before a reactor trip. Post-trip, the contents are released via the condenser air ejector that is aligned to the Unit No. 2 stack before loss of condenser vacuum. The radioactive gases are also released via the Main Steam Safety Valves (MSSVs) and Atmospheric Dump Valves (ADVs) directly to the atmosphere without holdup or decontamination.

According to the licensee's description of the updated SGTR thermal hydraulic analysis with a LOOP, the radioactive steam is released directly to the atmosphere via the MSSVs and ADVs. The updated SGTR analysis with a LOOP also postulates an increase in dose consequences compared to an SGTR without a LOOP.

The limits of radiation doses received by individuals as the result of fission products released during postulated accidents are given in 10 CFR Part 100, Reactor Site Criteria, and 10 CFR Part 50, Appendix A, GDC-19, Control Room. According to the licensee's application, the results of the new analysis are within the limits of 10 CFR Part 100 and GDC-19. The licensee's December 21, 2000, application included the following tables:

CASE 1 Iodine Spike Caused by Accident Calculated Doses (Rem)					
ORGAN	Exclusion Area Boundary (Old Value)	Exclusion Area Boundary (New Value)	Low Population Zone (Old Value)	Low Population Zone (New Value)	NRC DOSE CRITERIA (SRP 15.6.3)
Thyroid	0.160	15.4	0.017	2.1	30
Whole Body	0.146	2.2	0.045	0.3	2.5

CASE 2 Pre-accident Iodine Spike Calculated Doses (Rem)					
ORGAN	Exclusion Area Boundary (Old Value)	Exclusion Area Boundary (New Value)	Low Population Zone (Old Value)	Low Population Zone (New Value)	NRC DOSE CRITERIA (10 CFR 100 limits)
Thyroid	0.813	27.8	0.085	3.7	300
Whole Body	0.146	0.8	0.045	0.1	25

3.0 EVALUATION

The staff's evaluation consists of determining the acceptability of the changes to the assumptions used in the SGTR model, and then evaluating the results of the analysis, based on the new model, against the relevant regulatory criteria described above. In general, if the licensee uses an assumption that is more conservative than the assumption used in the current licensing basis, it is readily accepted as long as the assumption does not result in non-compliance with the regulatory criteria. However, if the licensee uses an assumption that is less conservative than the current licensing basis, this assumption requires a greater level of scrutiny because of the potential for either creating an erroneous margin of safety or exceeding regulatory limits.

For example, according to the proposed revisions to FSAR Section 14.6.3.5 in the licensee's December 21, 2000, application, in the event of an SGTR, there are five ways to cool and depressurize the isolated steam generator. The option that would result in the greatest offsite dose is the option chosen for the analysis in order to account for the worst case, whereas the option chosen during an actual event would be the most appropriate course of action based on the specific plant conditions. Therefore, conservative analysis assumptions lead to greater postulated offsite dose results. Consequently, if the licensee chose any of the other four ways, other than the option that yielded the most conservative results, to cool and depressurize the steam generator in their analysis, the actual method chosen could result in greater offsite doses than evaluated. This would be a non-conservative assumption and would require acceptable justification.

The licensee’s revised SGTR analysis uses the same RETRAN-02 Mod 3 computer code as that for the previous analysis. The use of RETRAN model for performing the MP2 SGTR analysis was previously reviewed and accepted by the staff. The SGTR analysis modeled the key emergency operating procedure (EOP) operator actions consistent with the Millstone Unit 2 EOPs. The significant changes to the methodology in the revised SGTR thermal-hydraulic analysis are discussed below:

The licensee's revised SGTR thermal-hydraulic analysis assumes that a LOOP and reactor trip occur at the time of the event initiation (instantaneous double-ended rupture of a steam generator tube). The loss of offsite power leads to a loss of forced RCS circulation which results in higher RCS temperature, higher fraction of the break flow flashing into the affected steam generator, and slower RCS cooldown and depressurization. The staff finds that the assumption of the reactor trip at the time of event initiation without time delay results in conservative dose calculations and is consistent with the assumption used in the licensee’s confirmatory analysis submitted on November 8, 1985. This analysis was reviewed and accepted by the staff in a letter dated January 21, 1987. Therefore, the staff finds this change acceptable.

The assumed LOOP will also cause the loss of condenser vacuum and instrument air. The ADVs are assumed to be inoperable for 30 minutes from the time of the event initiation and LOOP due to the loss of the instrument air supply. The ADVs at MP2 are not backed up by nitrogen bottles; a local manual operator action is credited for the operation of the ADVs for RCS cooldown. Consequently, the delayed RCS cooldown will prolong the duration of the break flow. The staff finds that these assumptions are acceptable because they result in a conservative dose calculation when considering the effects of a LOOP.

The revised analysis assumes higher high pressure safety injection (HPSI) flow to increase the primary to secondary break flow. The effect of this assumption on the thermal hydraulic analysis is independent of the effects of a LOOP following the SGTR. The staff finds that this assumption is acceptable because it results in a conservative dose calculation.

The revised analysis assumes lower auxiliary feedwater (AFW) flow and a lower steam generator water level for AFW actuation. This results in an increase of steam assumed to be released to the environment. The effect of this assumption on the thermal hydraulic analysis is independent of the effects of a LOOP following the SGTR. The staff finds that this assumption is acceptable because it results in a conservative dose calculation.

The revised analysis assumes continuous mass releases following the accident until RCS cooldown to shutdown cooling entry conditions. The effect of this assumption on the thermal hydraulic analysis is independent of the effects of a LOOP following the SGTR. The staff finds that this assumption is acceptable because it results in a conservative dose calculation.

The revised analysis assumes the lowest possible setpoint for the MSSVs on the ruptured steam generator and highest possible setpoint for the MSSVs on the intact steam generators. This will maximize the assumed mass release from the ruptured steam generator. The effect of this assumption on the thermal hydraulic analysis is independent of the effects of a LOOP following the SGTR. The staff finds that this assumption is acceptable because it results in a conservative dose calculation.

The revised analysis accounts for the iodine release associated with flashing of the primary to secondary break flow. As noted in Table 1 the licensee evaluated two cases to determine the consequences of the SGTR. The licensee stated in its June 29, 2001, supplement that the current MP2 licensing basis is to end the concurrent iodine spike at 4 hours. Dominion stated that the NRC approved this assumption in license Amendments Nos. 90 and 195, and that there is no significant increase in consequences if the spike is assumed to last more than 4 hours. While the staff's review of these amendments could not confirm that the staff had previously approved this specific assumption, the staff agrees that for the current analysis there is no significant increase in the consequences of an accident if the spike is assumed to last beyond 4 hours. Therefore, for the purpose of analyzing the SGTR using the current assumptions, the staff finds the assumption that the iodine spike ends at 4 hours is acceptable. However, if in the future the licensee changes assumptions which alter the release durations, the assumption that the spike ends at 4 hours may no longer be valid.

The revised analysis calculated thyroid doses using ICRP-30 dose conversion factors. Though ICRP-30 dose conversion factors are not as conservative as those used in the previous analysis, the staff acknowledged and accepted the use of ICRP-30 dose conversion factors in the safety evaluation associated with Amendment No. 228 to Facility Operating License No. DPR-65 for MP2 issued March 10, 1999. The staff finds the use of ICRP-30 dose conversion factors acceptable because they have already been approved by the staff and are already included in the licensing basis for MP2.

The revised analysis changed the air ejector partition factor shown in FSAR Table 14.6.3-5 to a value of 1.0. The licensee submitted changes to the air ejector partition factor given in Table 14.6.3-5. The staff requested justification for this change in a request for additional information. The licensee's response dated June 29, 2001, stated that the air ejector partition factor was not credited in the radiological evaluation. Therefore, this value was not reviewed since it does not pertain to the determination of the radiological consequences of this accident and does not have any impact on the health and safety of the public.

As described in the licensee's application, the revised analysis changed the values assumed for atmospheric dispersion (χ/Q) and breathing rate. The χ/Q values used in the SGTR analysis for the site boundary and control room intake (based on release from the atmospheric dump valves and main steam safety valves) are identical to the corresponding values used for the Main Steam Line Break (MSLB) analysis as reflected in FSAR Table 14.1.5.3-1. The MSLB analysis was approved by the NRC via Amendment No. 228 which was issued on March 10, 1999. The staff finds that the MSLB atmospheric dispersion factors are acceptable for use in the SGTR analysis based on the similarity between the release and intake points. The breathing rates used in the SGTR analysis are identical to the values stated in Regulatory Guide 1.4, Position 2.c, and therefore, are acceptable.

The staff reviewed the licensee's proposed changes to the SGTR accident described in its December 21, 2000, and June 29, 2001, submittals with regard to the applicable regulatory criteria, current FSAR analyses, regulatory guidance, and staff experience in reviewing similar analyses. The staff also performed independent calculations to confirm offsite and control room dose conclusions made by the licensee. The design inputs utilized by the staff to evaluate this accident are given in Tables 1 and 2. However, the staff's acceptance of this license amendment request is based on the licensee's docketed analysis, not the analyses performed by the staff to confirm the licensee's conclusions.

Based on the preceding evaluation, the staff finds reasonable assurance that an SGTR accident, concurrent with a LOOP, at MP2 will result in radiological consequences less than the dose guidelines of 10 CFR Part 100 and 10 CFR Part 50, Appendix A, GDC-19. Therefore, the staff finds that the licensee's proposed changes to the revised thermal-hydraulic analysis are acceptable. In addition, the staff reviewed the licensee's proposed changes to Sections 14.0.11 and 14.6.3 of the MP2 FSAR and finds that they are consistent with the revised analysis.

With regard to control room habitability, the staff is currently working toward resolution of generic issues related to control room habitability, including the validity of control room infiltration rates assumed by the licensee in analyses of control room habitability. Since the staff has not completed its work on the generic issues, the staff may require, through an appropriate regulatory process, additional activities that may impact the licensee's SGTR analysis.

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

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the *Federal Register* on April 12, 2002 (67 FR 18044). Accordingly, based upon the environmental assessment, the staff has determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

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.

Principal Contributors: W. Blumberg
C. Liang

Date: April 29, 2002

Table 1

MP2 Steam Generator Tube Rupture Parameters Used by Staff

Reactor power (2700 x 1.02 (Uncertainty in power measurements)), MWt	2,754
Primary to secondary leak rate per steam generator, gpm	0.035
Reactor coolant system maximum allowable concentration	
(Dose Equivalent Iodine-131, DEI)	1.0 μCi
Steam generator maximum allowable concentration (DEI)	0.1 μCi
Pre-accident spike iodine concentration (DEI)	60 μCi/gm
Melted fuel percentage (%)	0.0
Mass Released from SGTR Event, (lbm)	
Ruptured Steam Generator 0-2 hr. (ADVs and MSSVs)	219,883
Ruptured Steam Generator 2-16 hr. (ADVs and MSSVs)	0
Intact Steam Generator 0-2 hr. (ADVs and MSSVs)	670,856
Intact Steam Generator 2-16 hr. (ADVs and MSSVs)	2,014,036
Flashed Break Flow 0-2 hr.	5,240
Flashed Break Flow 2-16 hr.	0
Intact steam generator minimum mass (lbm)	100,000
Site boundary dispersion factors	See Table 2

Case 1: Concurrent iodine spike equivalent to 500 times the equilibrium iodine appearance rate at the Technical Specification limit.
Case 2: Pre-accident iodine spike assumed to be 60 μCi/gm.

Control Room (CR)

Type of control room design	Isolation with recirculation
Unfiltered intake (0-10 seconds), cfm	800
Unfiltered inleakage, intake isolated (10 seconds -30 days), cfm	130
Filtered recirculation (10 seconds -30 days), cfm	2250
Recirculation iodine filter efficiency, %	90
Free volume of the control room, ft^3	3.6E+4
Breathing rate, m^3/s	3.47×10^{-4}
Control room occupancy factor	
0-24 hrs	1.0
1-4 days	0.6
4-30 days	0.4

Other Parameters

Dose conversion factors	ICRP-30
Offsite Breathing rate, offsite,m^3/s	
0-8 hours	3.47×10^{-4}
8-24 hours	1.75×10^{-4}
>24 hours	2.32×10^{-4}
Atmospheric Dispersion Factors	Table 2

Table 2

MP2 Atmospheric Relative Concentration (χ/Q) Values
Used by Staff (in units of sec./m^3)

Time Period (hrs.)	Control Room	EAB	LPZ
0-2	3.19×10^{-3}	3.66×10^{-4}	4.80×10^{-5}
2-4	3.19×10^{-3}		4.80×10^{-5}
4-8	3.19×10^{-3}		2.31×10^{-5}
8-24	2.05×10^{-3}		1.60×10^{-5}
24-96	7.61×10^{-4}		7.25×10^{-6}
96-720	2.13×10^{-4}		2.32×10^{-6}