Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Rope Ferry Road
Waterford, CT 06385

Dominion

JUN 2 9 2001

Docket No. 50-336
B18424

RE: 10 CFR 50.90

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Millstone Nuclear Power Station, Unit No. 2
Response to Request for Additional Information
License Amendment Request - Unreviewed Safety Question
Proposed Revision to Final Safety Analysis Report (FSAR)
Radiological Consequences of Steam Generator Tube Rupture
(PLAR 2-00-4) (TAC NO. MB0866)

The purpose of this letter is to provide the Nuclear Regulatory Commission (NRC) with response to a Request for Additional Information[(1)] received from the NRC Staff regarding a proposed license amendment request. The proposed license amendment request (PLAR 2-00-4)[(2)] seeks approval for changes to the Millstone Nuclear Power Station, Unit No. 2 Final Safety Analysis Report (FSAR), Chapter 14, description of the steam generator tube rupture event and its associated radiological dose consequences. Attachment 1 contains responses to all twelve (12) questions.

There are no regulatory commitments contained within this letter.

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(1) J. I. Zimmerman, (NRC) letter to R. P. Necci, "Request for Additional Information (RAI) Regarding Revisions to the Millstone Nuclear Power Station, Unit No. 2 - Final Safety Analysis Report (TAC NO. MB0866)," dated June 4, 2001.

(2) R. P. Necci letter to the U.S. NRC, "Millstone Nuclear Power Station, Unit No. 2, License Amendment Request - Unreviewed Safety Question, Proposed Revision to Final Safety Analysis Report, Radiological Consequences of Steam Generator Tube Rupture (PLAR 2-00-4)," dated December 21, 2000.

A001

If you should have any questions on the above, please contact Mr. Ravi Joshi at 860-440-2080.

Very truly yours,

DOMINION NUCLEAR CONNECTICUT, INC.

J. Alan Price, Vice President
Nuclear Technical Services - Millstone

Sworn to and subscribed before me

this 29th day of June, 2001

Sandra J. Anton
Notary Public

My Commission expires SANDRA J. ANTON NOTARY PUBLIC COMMISSION EXPIRES MAY 31, 2005

Attachments (2)

cc: H. J. Miller, Region I Administrator
J. T. Harrison, NRC Project Manager, Millstone Unit No. 2
S. R. Jones, Senior Resident Inspector, Millstone Unit No. 2

Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

Docket No. 50-336
B18424

Attachment 1

Millstone Nuclear Power Station, Unit No. 2

License Amendment Request - Unreviewed Safety Question
Response to Request for Additional Information
Proposed Revision to Final Safety Analysis Report
Radiological Consequences of Steam Generator Tube Rupture
(PLAR 2-00-4)
Specific Responses to 12 Questions

License Amendment Request - Unreviewed Safety Question
Response to Request for Additional Information
Proposed Revision to Final Safety Analysis Report
Radiological Consequences of Steam Generator Tube Rupture (PLAR 2-00-4)
Specific Responses to 12 Questions

Question 1) *The proposed changes to FSAR Section 14.0.11 indicate that in the new analysis of a steam generator tube rupture (SGTR), a reactor trip is assumed to occur at the time of the event initiation without time delay. It is our understanding that the purpose of this assumption is to maximize the radiological consequences following the SGTR. Please provide a discussion to demonstrate why this assumption will lead to more conservative results as compared to a delayed reactor trip with respect to the radiological consequences.*

Response: If the reactor trip is modeled to occur on thermal margin low pressure (TMLP) low pressure automatic trip, there is a pre-trip component to the atmospheric releases. The pre-trip component of the activity will be less significant due to the releases going to the condenser and out the air ejector with an air ejector partition factor that reduces the activity. Furthermore, the air ejector releases would go through the Millstone elevated stack prior to the Enclosure Building Filtration Actuation Signal (EBFAS). Therefore, the pre-trip offsite dose component would be even smaller, due to the dispersion factor of the elevated release point. With the model that trips the reactor at the time of tube break, all of the releases are direct to the atmosphere and at ground level. Therefore, the model is intended to maximize the direct atmospheric releases.

Question 2) *You stated in your submittal that there are changes in the assumptions and methods used in the updated thermal hydraulic analysis of the SGTR accident. Please confirm that the computer codes used in the updated analysis are consistent with that for the current analysis and that they have been previously approved by the staff; also, please confirm that all changes of assumptions not addressed in your submittal will not lead to non-conservative results.*

Response: The computer code used to perform the latest SGTR analysis is the same RETRAN-02 Mod 3 code used in the previous SGTR analysis. The changes in the assumptions of the latest SGTR analysis, stated in the submittal as well as not addressed in the submittal, do not lead to non-conservative results.

The use of RETRAN model for performing the Millstone Unit No. 2 SGTR analysis was reviewed by the NRC in the 1983-1987 time frame. There were several rounds of requests for additional information and responses, including additional confirmatory calculations.

In a letter dated January 21, 1987, the Staff accepted the confirmatory calculation information and determined that it met the Staff's request for additional information to determine the acceptability of the Millstone Unit No. 2 Cycle 6 Reload Analysis.

It should also be noted that the Northeast Utilities (NU) non-LOCA (Loss Of Coolant Accident) analysis methodology, including SGTR, was approved by the NRC on July 26, 1988, for application to the Haddam Neck Plant. The model benchmarking provided for this methodology included a comparison to an actual Millstone Unit No. 2 loss of load transient as well as Haddam Neck Plant transients.

Question 3) *You stated in your submittal that the atmospheric dump values (ADVs) are assumed to not be available for 30 minutes from the time of reactor trip due to loss of instrument air. Are there any safety grade nitrogen bottles installed to support the operation of ADVs? Per the emergency operating procedures (EOPS) at Millstone Unit 2, the ADVs may be opened much earlier than 30 minutes into the event. Please provide a discussion to demonstrate that the assumed late actuation of ADVs would lead to more limiting results in radiological consequences.*

Response: The ADVs at Millstone Unit No. 2 are air operated valves. There are no nitrogen bottles installed to provide backup air to the ADVs.

The operator action assumption of 30 minutes to locally operate the ADVs is chosen to be conservative with regard to the offsite dose consequences. Once the ADVs are open, the duration to reach the ruptured steam generator isolation criteria of hotleg temperature less than 515°F would be comparable for whether the ADVs are opened with 30 minute delay or shorter. The conservatism is introduced by the assumed delay of the manual operator action resulting in a longer duration for the primary to secondary break flow, and more activity in the steam generator prior to opening the ADVs. For the concurrent iodine spike, the longer delay also leads to more activity. These two components would make the offsite dose consequences worse. Any decay effects of the iodine over the duration that is less than 30 minutes (e.g. 15 minutes) is considered not significant.

Question 4) *Please confirm that the event scenario of the updated SGTR analysis is consistent with the operator actions per EOPs at MP2 except for the assumptions for the time of reactor trip and ADV initiation.*

Response: The SGTR analysis modeled the key Emergency Operating Procedure (EOP) operator actions. These mode assumptions are consistent with the Millstone Unit No. 2 EOPs.

Question 5) Please expand the sequence of events for the SGTR accident contained in the proposed revision of FSAR Table 14.6.3-3 (Attachment 3, Insert B of the application) to include all significant changes to primary and secondary system parameters and all major plant system responses during the postulated SGTR. This information is needed to assist the staff in assessing the radiological consequences for the event.

Response: The "Sequence of Events for the Steam Generator Tube Rupture Event" Table, that was submitted as part of this license amendment request, has been expanded to include applicable information found in the current FSAR Table 14.6.3-4. The new table is included as part of Attachment 2 to this RAI response.

Question 6) Please provide the atmospheric dispersion factors from the release points in the accident to the control room. If the Nuclear Regulatory Commission (NRC) staff has not reviewed and approved these values, provide the information necessary to calculate these values. If the NRC has previously approved these atmospheric dispersion factors, please provide the information necessary to verify that the NRC staff has performed a review of the values calculated.

Response: The control room intake dispersion factors (sec/m^3) from the release points, Main Steam Safety Valve/Atmospheric Dump valves listed in the Millstone No. 2 Final Safety Analysis Report (FSAR) Table 14.1.5.3-1 are as follows:

0 - 8 hrs	3.19E-03 sec/m^3
8 - 24 hrs	2.05E-03 sec/m^3
24 - 96 hrs	7.61E-04 sec/m^3
96 - 720 hrs	2.13E-04 sec/m^3

These factors were used for the SGTR dose calculation for the control room and includes the occupancy factors.

These X/Q's were previously used in the main steam line break (MSLB) analysis. The MSLB analysis was reviewed and approved by the NRC via License Amendment No. 228.[(1)]

(1) S. Dembek letter to M. L. Bowling, "Issuance of Amendment - Millstone Nuclear Power Station, Unit No. 2, (TAC NOS. MA3410 and MA3672)," dated March 10, 1999.

Question 7) Please provide details on when the control room ventilation intake is isolated and is set to emergency intake mode following a SGTR.

Response: The control room post accident ventilation design corresponds to an isolation type design whereby on receipt of an accident signal (Safety Injection (SI) or control room intake radiation monitor high alarm signal), the control room isolates within 5 seconds. The maximum incoming outside fresh air prior to control room isolation is 800 cfm. Subsequent to isolation, inleakage is no greater than 130 cfm. The Control Room Emergency Ventilation (CREV) cleanup system is initiated via operator action, 10 minutes after the accident.

The intake monitor response time is determined by the final count rate at the detector, the monitor setpoint, the background count rate and the time constant of the microprocessor.

The detector response function is 1 mR/hr per 9E-03 uCi/cc Xe-133 dose equivalent. The monitor setpoint is 1 mR/hr. The initial Xe-133 DE concentrations at the control room intake for the design basis SGTR are calculated as follows:

RSC Xe-133 DE Concentration			
Isotope	Tech Spec Conc.	Energy / dis	Conc. * E/dis
Kr-85m	4.8	0.1576	0.75648
Kr-85	11	0.002231	0.024541
Kr-87	4.5	0.7825	3.52125
Kr-88	8.3	1.935	16.0605
Xe-131m	21	0.02008	0.42168
Xe-133m	2.1	0.04146	0.087066
Xe-133	75	0.04519	3.38925
Xe-135m	3.9	0.4307	1.67973
Xe-135	25	0.2465	6.1625
Xe-138	3.6	1.096	3.9456
Total			36.048597
RCS Xe-133 DE = 36.05 / 0.04519 = 800 uCi/gm			

CR Intake Conc. = (RCS Activity) x RCS Leak Rate x (X/Q)
= 800 uCi/gm x (22,461 lb/409.3 sec) x 2.92E-3 s/m3
= 0.0582 uCi/cc

Dose Rate = 0.0582 uCi/cc / 0.009 uCi/cc / (mR/hr) = 6.5 mR/hr

The step increase of the instantaneous dose rate to the detector at the arrival of the plume is 6.5 mR/hr. The time constant of the monitor response for an exposure range of 1 to 10 mR/hr is 0.033 minutes. The monitor response time is therefore 0.33 seconds.

The monitor response time will be assumed to be 1 second. The control room isolation delay time equals the monitor response time (1 sec) plus the damper closing time (5 sec) or 6 seconds. For this analysis, 10 seconds is used.

The control room model is summarized below:

0 - 10 seconds:	800 cfm unfiltered intake
10 sec - 10 min:	130 cfm unfiltered inleakage (ventilation intake is isolated)
10 min - 30 days:	130 cfm unfiltered inleakage, 2250 cfm recirculation with iodine filtering efficiency of 90%.

Question 8) Please provide the doses calculated for the SGTR control room dose analyses.

Response: The calculated doses to the Millstone Unit No. 2 control room for a steam generator tube rupture (SGTR) event with a loss of offsite power are provided below. The results show that these consequences (dose) are bounded by the MSLB and the loss-of-coolant-accident (LOCA) dose consequences. These doses are listed in the FSAR Tables 14.1.5.3-2, 14.1.5.3-3 and 14.8.4-5.

<u>Control Room Doses for the SGTR Event with Loss of Power (LOP)</u>

Preaccident Iodine Spike:	Thyroid	15.55 rem
	Whole Body	0.08 rem
	Beta	1.62 rem
Concurrent Iodine Spike:	Thyroid	8.09 rem
	Whole Body	0.11 rem
	Beta	1.88 rem

Question 9) Please justify ending the iodine spike at four (4) hours following the accident. What mechanism stops the spike at this time?

Response: The Millstone Unit No. 2 current licensing basis for the SGTR event is to end the iodine spike at four (4) hours after the accident.

The Millstone Unit No. 2 FSAR (current) Section 14.6.3.6.2 and Tables 14.6.3-5 and 14.6.3-6 provide radiological consequences of the SGTR based on the assumption that the iodine spike ends at four (4) hours. The NRC approval of this analysis is documented in License Amendment No. 90[2] and

[2] J. R. Miller letter to W. G. Counsil, "Issuance of Amendment No. 90," dated December 30, 1983.

No. 195.[3] The analysis presently under review by the NRC uses the same assumption used in the previous analyses.

In addition, there is no significant increase in consequences if the spike is assumed to last beyond 4 hours. The SGTR analysis assumes that the ruptured steam generator is isolated within 2 hours. Therefore, the only contribution after 4 hours will be from the intact steam generator which has a primary to secondary leakage limit of 0.035 gpm.

Question 10) *Please provide the activity of each radionuclide used in the analysis for an equilibrium reactor coolant system concentration of 1 micro-Ci/gm Dose Equivalent Iodine-131. What inputs and assumptions were used to calculate these values?*

Response: The activity of each radionuclide used in the analysis for an equilibrium reactor coolant system concentration of 1 micro-Ci/gm Dose Equivalent Iodine-131 is provided below:

I-131:	5.5E-01
I-132:	2.4E+00
I-133:	1.7E+00
I-134:	3.9E+00
I-135:	3.1E+00
Kr-85:	1.1E+01
Kr-85m:	4.8E+00
Kr-87:	4.5E+00
Kr-88:	8.3E+00
Xe-131m:	2.1E+01
Xe-133:	7.5E+01
Xe-133m:	2.1E+00
Xe-135:	2.5E+01
Xe-135m:	3.9E+00
Xe-138:	3.6E+00

Details on the analytical model may be found in ANSI/ANS-18.1-1984, "American National Standard - Radioactive Source Term for Normal Operation of Light Water Reactors." The starting point of the analysis was the radionuclide coolant concentrations specified in the standard for a reference Pressurized Water Reactor (PWR) with U-tube steam generators. These concentrations were adjusted for Millstone Unit No. 2 specific parameters and prepared in accordance with the ANSI/ANS model to reflect the operating differences between MP2 and the reference plant.

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[3] G. S. Vissing letter to R. E. Busch, "Issuance of Amendment No. 195 (TAC No. MA93794)," dated January 18, 1996.

These Millstone Unit No. 2 specific parameters are listed in the following table.

Thermal Power	2,754 MWt
Steam flow rate	11,802,957 lbm/hr
Nominal volume of water & steam in RCS	10,981 ft3
Volume of Pressurizer	1,500 ft3
Nominal volume of water in pressurizer	975 ft3
Volume of steam in pressurizer	525 ft3
Volume of water in RCS	10,456 ft3
RCS Normal operating temperatures	Hot leg: 604 degrees F Cold Leg: 550 degrees F Avg. 577 degrees F
Specific volume of water in RCS	0.022676 ft3/lbm
Mass of water in RCS	461,105 lbm
Mass of water in each of the 2 S/Gs	132,257 lbm
Reactor coolant purification letdown flow rate	40-128 gpm through purification, 4 gpm through RCP seals bleedoff
Reactor Coolant letdown flow rate (yearly average for boron control)	500 lbm/hr
Steam generator blowdown flow rate (total)	540 gpm
Fraction of radioactivity in blowdown stream that is not returned to the secondary coolant system	100%
Flow through the purification system cation demineralizer	40 - 128 gpm
Ratio of condensate demineralizer flow rate to the total steam flow rate	1:1
Fraction of the noble gas activity in the letdown stream that is not returned to the reactor coolant system	0% when the degassifier is not operating
RCS Tech Spec limit Halogens Noble Gases	Initial RCS iodine activities correspond to 1uCi/gm DE-131 100/E avg. of noniodines
Secondary side tech spec limits	0.1 uCi/gm DE-131

Question 11) Please provide the release rates of each radio nuclide used for the accident initiated spike source term and the assumptions and inputs used to calculate these values.

Response: The following equation was used to generate the appearance rates:

$P = C * S * WP * A * (R + \lambda)$

where:
- P = appearance rate (uCi/sec) of radioiodine in the RCS
- C = conversion constant (453.59 gm/lbm per 3600 sec/hr)
- S = spiking factor
- WP = mass of water in RCS (4.611E+05 lbm)
- A = iodine activity in the RCS at the TS limit
- R = RCS cleanup rate (6.33E-02/hr)
- λ = radionuclide decay constant

Computation of Iodine Spike Appearance Rates				
Nuclide	RCS Conc. at TS Limit (A)	Decay Constant(λ)	Appearance Rate (P)	
	(uCi/gm)	(per hour)	(At TS limit)	(500 x spiked)
I-131	5.48E-01	3.592E-03	2.13E+03	1.07E+06
I-132	2.45E+00	3.014E-01	5.19E+04	2.59E+07
I-133	1.68E+00	3.332E-02	9.42E+03	4.71E+06
I-134	3.94E+00	7.907E-01	1.96E+05	9.78E+07
I-135	3.07E+00	1.049E-01	3.00E+04	1.50E+07
Total	1.17E+01		2.89E+05	1.45E+08

Question 12) Please provide the bases for the air ejector partition factor.

Response: The SGTR analysis presented in the proposed revision to the Millstone Unit No. 2 FSAR Section 14.6.3, assumes that the reactor trip and turbine trip will occur at the same time and a loss of offsite power is also lost along with the condenser. Therefore, the air ejector is not available as a release path. FSAR Table 14.6.3-3 shows the sequence of events. The revised FSAR Table 14.6.3-5, is included as part of Attachment 2. Note that the air ejector partition factor is not credited for the radiological evaluation.

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Attachment 2

Millstone Nuclear Power Station, Unit No. 2

License Amendment Request - Unreviewed Safety Questions
Response to Request for Additional Information
Proposed Revision to Final Safety Analysis Report
Radiological Consequences of Steam Generator Tube Rupture
(PLAR 2-00-4)
FSAR Marked Up Pages

00-MP2-47

2 TABLE 14.6.3-3

KEY PARAMETERS ASSUMED IN THE STEAM GENERATOR TUBE RUPTURE EVENT

Parameter	Units	Value
Initial Core Power Level	MWt	2754
Core Inlet Temperature	°F	551
RCS Pressure	psia	2300
Initial Steam Generator Pressure	psia	933
Plugged U-Tubes SG1/SG2	#	1300/1200
Moderator Temperature Coefficient (ARO)	$x10^{-4}\ \Delta p$	-2.5
Doppler Multiplier		1.15

93-42 (6/94)

93-42

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00-MP2-47

INSERT B

SEQUENCE OF EVENTS FOR THE STEAM GENERATOR TUBE RUPTURE EVENT

TIME [1] (sec)	EVENT	SETPOINT OR VALUE
0	Tube Rupture Occurs	
0	Reactor Trip	
0	Concurrent Loss of Offsite Power; Loss of Forced Circulation	
0	Loss of Instrument Air; Loss of ADV auto actuation	
1	Ruptured SG MSSVs Begin to Lift (MSSVs Cycle - See Figure 14.6.3-8)	970 psia
3	Intact SG MSSVs Begin to Lift (MSSVs Cycle - See Figure 14.6.3-8)	1030 psia
9	Maximum SG Pressure Reached - Ruptured SG - Intact SG	 1054 psia 1075 psia
37	AFW Actuation Condition Reached on Low SG Level	10% Narrow Range
277	AFW Delivery Starts	
518	Pressurizer Empties	
1015	Intact SG MSSVs Close for Final Time	940 psia
1150	SI Flow Begins to Enter RCS	
1411	Ruptured SG MSSVs Close for Final Time	880 psia
1800 [2]	RCS Cooldown to T_{HOT} < 515°F Initiated Using ADVs	
3020	Pressurizer Begins to Refill	
3589	T_{HOT} < 515°F Achieved; Ruptured Steam Generator Isolated	

Notes:
1. Time values are rounded to the nearest second.
2. This is an assumed analytical time and is not a required operator action time.

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00-MP2-47

TABLE 14.6.3-5

(5/96)

ASSUMPTIONS FOR THE RADIOLOGICAL EVALUATION FOR THE STEAM GENERATOR TUBE RUPTURE ACCIDENT

Conservative Assumption	Basis
1) Reactor Coolant System Maximum Allowable Concentration I-131 (DEQ) = 1.0uCi/gm	Tech Specs
2) Steam Generator Maximum Allowable Concentration I-131 (DEQ) = .1uCi/gm	Tech Specs
3) Reactor Coolant System Maximum Allowable Concentration of Noble Gases Xe-133 (DEQ) = 100/E uCi/gm	Tech Specs
4) Steam Generator Partition Factor = .01	
5) Air Ejector Partition Factor - .0005	
6) Atmospheric Dispersion Coefficient:	95% maximum X/Q's for the years 1974-1977

Admin (9/96)

Receptor Location	Elevated (sec/m^3)	Ground Level (sec/m^3)
Site Boundary	1.03×10^{-4}	5.41×10^{-4}
LPZ	3.41×10^{-5}	5.55×10^{-5}

Conservative Assumption	Basis
7) Breather Rate = 3.47×10^{-4} m^3/sec	SRP 15.6.3
8) I-131 dose conversion factor = 1.49×10^{6} rem Ci	Reg. Guide 1.109- Adult-Thyroid Inhalation
9) Iodine Spiking Factors	NRC Criterion

a) Case 1: Concurrent iodine spike equivalent to 500 times equilibrium iodine appearance rate at Technical Specification limit.

b) Case 2: Preaccident iodine spike concentration based upon 60 times Technical Specification limit.

OO-MP2-47

INSERT D

ASSUMPTIONS FOR THE RADIOLOGICAL EVALUATION FOR THE STEAM GENERATOR TUBE RUPTURE EVENT

	Conservative Assumption		Basis
1)	Reactor Coolant System Maximum Allowable Concentration I-131 (DEQ) = 1.0 μCi/gm		Technical Specifications
2)	Steam Generator Maximum Allowable Concentration I-131 (DEQ) = .1 μCi/gm		Technical Specifications
3)	Reactor Coolant System Maximum Allowable Concentration of Noble Gases (DEQ) = $100/E_{bar}$ μCi/gm		Technical Specifications
4)	Steam Generator Partition Factor		SRP 15.6.3
	Iodine	0.01	
	Noble Gases	1.0	
5)	Air Ejector Partition Factor (not credited)		NUREG 0017
	Iodine	0.15	
	Noble Gases	1.0	
6)	Atmospheric Dispersion Coefficient		95% maximum X/Q's for the years 1974-1981
	Receptor Location	Ground Level (sec/m^3)	
	EAB	3.66×10^{-4}	
	LPZ 0-4 hr	4.80×10^{-5}	
	4-8 hr	2.31×10^{-5}	
	8-24 hr	1.60×10^{-5}	
	24-96 hr	7.25×10^{-6}	
	96-720 hr	2.32×10^{-6}	
7)	Breathing Rate		Reg. Guide 1.4
	0-8 hr	3.47×10^{-4} m^3/sec	
	8-24 hr	1.75×10^{-4} m^3/sec	
	24-720 hr	2.32×10^{-4} m^3/sec	
8)	Dose Conversion factor		ICRP 30
	I-131	1.073×10^{6} rem Ci	
	I-132	6.290×10^{3} rem Ci	
	I-133	1.813×10^{5} rem Ci	
	I-134	1.073×10^{3} rem Ci	
	I-135	3.145×10^{4} rem Ci	
9)	Iodine Spiking Factors a) Case 1: Concurrent iodine spike equivalent to 500 times equilibrium iodine appearance rate at Technical Specification limit. b) Case 2: Preaccident iodine spike concentration based upon 60 times Technical Specification limit.		SRP 15.6.3