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Dominion

May 27, 2004

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

Serial No.	04-285
MPS Lic/MAE	R0
Docket No.	50-423
License No.	NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3
PROPOSED TECHNICAL SPECIFICATION CHANGES
IMPLEMENTATION OF ALTERNATE SOURCE TERM

Pursuant to 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC) requests amendments in the form of changes to the Technical Specifications to Facility Operating License Number NPF-49 for Millstone Power Station Unit 3. The proposed changes are being requested based on the radiological dose analysis margins obtained by using an alternate source term consistent with 10 CFR 50.67. A discussion of the proposed Technical Specifications changes is provided in Attachment 1. The marked-up and proposed Technical Specifications pages are provided in Attachments 2 and 3, respectively. The associated Bases changes are provided in Attachment 6 for information only and will be implemented in accordance with the Technical Specification Bases Control Program and 10 CFR 50.59.

We have evaluated the proposed technical specifications changes and have determined that they do not involve a significant hazards consideration as defined in 10 CFR 50.92. The basis for our determination that the changes do not involve a significant hazards consideration is provided in Attachment 4. We have also determined that operation with the proposed changes will not result in any significant increase in the amount of effluents that may be released offsite and no significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed changes. The basis for our determination that the changes do not involve any significant increase in effluents or radiation exposure is provided in Attachment 5.

DNC requests approval of the approved change and implementation within 90 days upon issuance of the amendment.

The Site Operations Review Committee and Management Safety Review Committee have reviewed and concurred with the determinations.

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In accordance with 10 CFR 50.91(b), a copy of this License Amendment Request is being provided to the State of Connecticut.

There are no regulatory commitments contained in this letter.

If you have any questions or require additional information, please contact Mr. Paul R. Willoughby at (804) 273-3572.

Very truly yours,

Eugene S. Grecheck
Vice President – Nuclear Support Services

Attachments: (6)

Commitments made in this letter: None.

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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Eugene S. Grecheck, who is Vice President – Nuclear Support Services, of Dominion Nuclear Connecticut, Inc. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 27th day of May, 2004.

My Commission Expires: 3/31/08.

Maggie McClure

Notary

ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATION CHANGES
IMPLEMENTATION OF ALTERNATE SOURCE TERM

DISCUSSION OF CHANGES

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

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1.0 Introduction & Background

1.1 Introduction

This report describes the evaluations conducted to assess the radiological consequences of fully implementing the Regulatory Guide 1.183 (RG 1.183) (Reference 1) accident methodology for Millstone Unit 3. The accident source term discussed in Reference 1 is herein referred to as the Alternative Source Term (AST).

The evaluations documented herein have employed the detailed methodology contained in RG 1.183 for use in design basis accident analyses for alternative source terms. The results have been compared with the acceptance criteria contained either in 10 CFR 50.67 (Reference 2) or the supplemental guidance in RG 1.183.

This application, if granted, would:

- Implement RG 1.183 as the design basis source term for Millstone Unit 3,
- Allow Millstone Unit 3 to achieve a consistent design basis for all accident dose assessments,
- Increase operational flexibility by allowing increased Refueling Water Storage Tank (RWST) leakage,
- Increase operational flexibility by allowing increased unfiltered control room inleakage
- Increase operational flexibility by allowing increased unfiltered containment leakage and
- Remove from the Technical Specifications the Control Room Envelope Pressurization System.

All the radiological dose analyses for the above accidents were performed with a controlled version of the computer code RADTRAD-NAI 1.1 (QA) (Reference 3). The RADTRAD computer code calculates the control room and offsite doses resulting from releases of radioactive isotopes based on user supplied atmospheric dispersion factors, breathing rates, occupancy factors and dose conversion factors. Innovative Technology

Solutions of Albuquerque, New Mexico developed the RADTRAD code for the NRC. The original version of the NRC RADTRAD code was documented in NUREG/CR-6604 [Reference 4]. The Numerical Applications, Inc. (NAI) version of RADTRAD was originally derived from NRC/ITS RADTRAD, version 3.01. Subsequently, RADTRAD-NAI was changed to conform to NRC/ITS RADTRAD, Version 3.02 with additional modifications to improve usability. The RADTRAD-NAI code is maintained under NAI's QA program, which conforms to the requirements of 10CFR50, Appendix B.

Control Room Atmospheric Dispersion Factors were evaluated using the ARCON96 computer code (Reference 5). The ORIGEN and QADS computer codes from the SCALE code package (Reference 6) were used to evaluate the containment and filter shine doses.

1.2 Current Licensing Basis Summary

The current design basis radiological analyses that appear in the Millstone Unit 3 Updated Final Safety Analysis Report (UFSAR) consist of assessments of the following events:

1) Main Steam Line Break
2) Locked Rotor Accident
3) Rod Control Cluster Assembly (RCCA) Ejection Accident
4) Small Line LOCA Outside Containment
5) Steam Generator Tube Rupture
6) Loss of Coolant Accident
7) Fuel Handling Accident
8) Waste Gas System Failure
9) Radioactive Liquid Waste System Leak or Failure (Atmospheric Release)

1.3 Analysis Assumptions & Key Parameter Values

1.3.1 Selection of Events Requiring Reanalysis

In accordance with Standard Review Plan (SRP) 15.0.1, Section 1, Item Number 4, the following radiological analyses have been superceded:

a) Steam System Piping Failures Inside and Outside of Containment
b) Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break
c) Spectrum of Rod Ejection Accidents
d) Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment
e) Radiological Consequences of Steam Generator Tube Failure
f) Loss-of-Coolant Accidents
g) Radiological Consequences of Fuel Handling Accidents

A full implementation of the AST (as defined in Section 1.2.1 of Reference 1) is proposed for Millstone Unit 3. To support the licensing and plant operation changes discussed in Section 2.0 of this application, the following accidents were reanalyzed employing the RG 1.183 source term:

- Loss of Coolant Accident (LOCA),
- Fuel Handling Accident (FHA),
- Main Steam Line Break (MSLB) Accident,
- Locked Rotor Accident (LRA),
- Rod Control Cluster Assembly (RCCA) Ejection Accident (REA) and
- Steam Generator Tube Rupture (SGTR) Accident.

The analysis methodology applied the guidance of RG 1.183, in conjunction with the Total Effective Dose Equivalent (TEDE) methodology. If this request is granted, the implementation of RG-1.183 in this plant-specific application will become the bases for the source term employed in design basis radiological analyses for Millstone Unit 3.

The discussion of radiological consequences for "Failure of Small Lines Carrying Primary Coolant Outside Containment" are not required and will be deleted from the FSAR. The "Waste Gas System Failure" and "Radioactive Liquid Waste System Leak or Failure (Atmospheric Release)" radiological analyses are being retained for FSAR Chapter 11, Radioactive Waste Management. These events are unaffected by the conversion to AST. The FSAR thyroid and whole body results for these events as shown in Table 15.0-8 will be converted to TEDE so as to employ consistent methodology. Both events result in only an EAB dose. The TEDE result is:

1) Waste Gas System Failure: 0.22 rem
2) Radioactive Liquid Waste System Leak or Failure (Atmospheric Release): 0.013 rem

The proposed licensing and plant operational changes are discussed in Section 2.0. These changes require appropriate changes to the Millstone Unit 3 Technical Specifications, which are also described in Section 2.0 of this report. The key changes considered are listed below:

a. revise definition of Dose Equivalent 1-131 in Section 1.10 of the Technical Specifications Definitions to reference Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, " 1989, as the source of thyroid dose conversion factors (Reference 8).
b. change Technical Specification 3/4.7.7, "Control Room Emergency Air Filtration System," Surveillance Requirements c.2 and d to reflect a methyl penetration less than or equal to 5% for the Control Room Emergency Air Filtration System filters instead of 2.5%.

c. delete Technical Specification 3/4.7.8, "Control Room Envelope Pressurization System." The Control Room Envelope Pressurization System is no longer credited in the accident analyses described in the evaluation.
d. change the leakage rate acceptance criteria for all penetrations that are Secondary Containment bypass leakage paths in Technical Specification Section 6.8.4.f, "Containment Leakage Testing Program," from ≤ 0.042 L_a to ≤ 0.06 L_a.
e. revise the bases Sections 3/4.7.7 and 3/4.7.8 to reflect the above listed changes in accordance with the Millstone Unit 3 Bases Control Program as described in Section 6.18 of the Technical Specifications.

It can be concluded from the evaluation summarized above that implementing the AST, in conjunction with the proposed plant operational changes, will require reanalysis of the LOCA, FHA, SGTR, MSLB, LRA, and REA. Sections 3.1 through 3.6, respectively, provide the detailed description of the re-analyses for these events.

1.3.2 Analysis Assumptions & Key Parameter Values

This section describes the general analysis approach and presents analysis assumptions and key parameter values that are common to the accident analyses performed to implement the RG 1.183 source term. Sections 3.1 through 3.6 of this Attachment provide specific assumptions that were employed for the LOCA, FHA, SGTR, MSLB, LRA and REA, respectively.

The dose analyses documented in this application employ the Total Effective Dose Equivalent (TEDE) calculation method as specified in RG-1.183 for AST applications. The Total Effective Dose Equivalent (TEDE) is determined at the Exclusion Area Boundary (EAB) for the worst 2-hour interval. TEDE for individuals at the Low Population Zone (LPZ) and for the Millstone Unit 3 Control Room personnel are calculated for the assumed 30-day duration of the event.

The TEDE concept is defined to be the Deep Dose Equivalent, DDE, (from external exposure) plus the Committed Effective Dose Equivalent, CEDE, (from internal exposure). In this manner, TEDE assesses the impact of all relevant nuclides upon all body organs, in contrast with the previous single, critical organ (thyroid) concept for assessing internal exposure. The DDE is nominally equivalent to the Effective Dose Equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE is used in lieu of DDE in determining the contribution of external dose to the TEDE. EDE dose conversion factors were taken from Table III.1 of Federal Guidance Report 12 (Reference 9) per Section 4.1.4 of Reference 1.

There are a number of analysis assumptions and plant features that are used in the analysis of all of the events. These items are presented in Table 1.3-1 through 1.3-5.

Table 1.3-1

Control Room Assumptions & Key Parameters Employed in the AST Analyses

Assumption / Parameter	Value
Control Room Effective Volume	2.38E+05 ft3
Normal Control Room Intake Flow Rate prior to Isolation	1595 cfm
Unfiltered Inleakage during Periods Of Neutral Pressure	350 cfm
Unfiltered Inleakage during Periods Of Positive Pressure	100 cfm
Emergency Ventilation System Recirculation Flow Rate	666 cfm
Emergency Ventilation System Pressurization Flow Rate	230 cfm
Response Time for Control Room Inlet Radiation Monitor to generate the Control Building Isolation (CBI) Signal (Note: this value is validated for each accident analysis in the referenced calculations)	5 seconds
Response Time for Control Room to Isolate upon Receipt of CBI	5 seconds
Time credited for delay of Control Room Envelope Pressurization System	1 minute
Time credited for Control Room Envelope Pressurization System Discharge to the Control Room	60 minutes
Time credited for operator action to align Control Room Emergency Ventilation System after the Control Room Envelope Pressurization System stops	40 minutes
Time to place Emergency Ventilation System in service (summation of the 3 preceding time credits)	101 minutes after CBI signal
Filter Efficiencies	90% elemental & aerosol 70% organic
Millstone Unit 3 Control Building Wall Thickness:	2 feet concrete
Millstone Unit 3 Control Room Ceiling Thickness:	8 inches concrete
Millstone Unit 3 Control Building Roof Thickness:	1ft-10in concrete

Table 1.3-1

Control Room Assumptions & Key Parameters Employed in the AST Analyses

Assumption / Parameter	Value
Millstone Unit 3 Control Room Occupancy Factors	
0 – 24 hours	1.0
24 – 96 hours	0.6
96 – 720 hours	0.4

Table 1.3-2

NSS Assumptions & Key Parameters Commonly Employed in the AST Analyses

Assumption / Parameter	Value
Containment Free Volume	2.35E6 ft^3
Millstone Unit 3 Containment Wall Thickness:	4.5ft concrete
Millstone Unit 3 Containment Dome Thickness:	2.5ft concrete
Distance from Millstone Unit 3 Containment to the MP3 Control Room:	228ft
Millstone Unit 3 Containment Inner Radius:	70ft

Table 1.3-3

Offsite Atmospheric Dispersion Factors (sec/m3)

Receptor/ Source Location / Duration	X/Q (sec/m^3)
Exclusion Area Boundary (EAB) (0 – 720 hours)	
Containment	5.42E-04
Millstone Stack (includes fumigation)	1.00E-04
Other Release Points	4.30E-04
Low Population Zone (LPZ)	
Non-Millstone Stack Release Points	
0 – 8 hours	2.91E-05
8 – 24 hours	1.99E-05
24 – 96 hours	8.66E-06
96 – 720 hours	2.63E-06
Millstone Stack (includes fumigation)	
0 – 4 hours	2.69E-05
4 – 8 hours	1.07E-05
8 – 24 hours	6.72E-06
24 – 96 hours	2.46E-06
96 – 720 hours	5.83E-07

Table 1.3-4

Control Room Atmospheric Dispersion Factors

Source Location / Duration		X/Q (sec/m^3)
Turbine Building Ventilation Vent		
	0 – 2 hour	2.82E-03
	2 – 8 hour	1.65E-03
	8 – 24 hour	6.67E-04
	24 – 96 hour	4.83E-04
	96 – 720 hour	3.80E-04
Main Steam Valve Building Ventilation Exhaust		
	0 – 2 hour	1.46E-03
	2 – 8 hour	8.76E-04
	8 – 24 hour	3.42E-04
	24 – 96 hour	2.71E-04
	96 – 720 hour	1.96E-04
Containment Enclosure Building		
	0 – 2 hour	5.34E-04
	2 – 8 hour	3.23E-04
	8 – 24 hour	1.38E-04
	24 – 96 hour	8.78E-05
	96 – 720 hour	7.42E-05
Engineering Safety Features Building Ventilation Exhaust		
	0 – 2 hour	3.18E-04
	2 – 8 hour	2.26E-04
	8 – 24 hour	9.06E-05
	24 – 96 hour	6.42E-05
	96 – 720 hour	4.59E-05
Refueling Water Storage Tank Vent		
	0 – 2 hour	2.61E-04
	2 – 8 hour	1.59E-04
	8 – 24 hour	6.45E-05
	24 – 96 hour	4.83E-05
	96 – 720 hour	3.63E-05
Millstone Stack		
	0 – 4 hour	1.39E-04
	4 – 8 hour	3.23E-05
	8 – 24 hour	1.56E-05
	24 – 96 hour	3.20E-06
	96 – 720 hour	3.30E-07

Table 1.3-4

Control Room Atmospheric Dispersion Factors

Turbine Building		
	0 – 2 hour	5.40E-03
	2 – 8 hour	3.51E-03
	8 – 24 hour	1.38E-03
	24 – 96 hour	1.01E-03
	96 – 720 hour	8.49E-04

Table 1.3-5

Breathing Rates

Source Location / Duration		X/Q (m^3/sec)
Offsite (EAB & LPZ)		
	0 – 8 hour	3.50E-04
	8 – 24 hour	1.80E-04
	24 – 720 hour	2.30E-04
Control Room		
	0 – 720 hour	3.50E-04

2.0 Proposed Licensing Basis Changes

This section provides a summary description of the key proposed licensing basis changes that are justified with the Millstone Unit 3 AST analyses accompanying this license amendment request.

2.1 Implementation of Regulatory Guide 1.183 Methodology as Design Basis Source Term

This report supports a request to revise the design basis accident source term for Millstone Unit 3. Subsequent to approval of this license amendment, the design basis source term for use in evaluating the consequences of design basis accidents will become the source term documented in RG 1.183 (Reference 1), including any deviations approved by the NRC staff. This license amendment application is made pursuant to the requirements of CFR § 50.67(b)(1), which specifies that any licensee seeking to revise its current accident source term used in design basis radiological consequences analysis shall apply for a license amendment.

2.2 Relaxation of Surveillance Requirements for the Control Room Emergency Air Filtration System Filter Efficiency

The Control Room Emergency Ventilation System is considered operable based on limiting the radiation exposure to personnel occupying the Control Room to the applicable dose. The proposed changes have been analyzed for radiological events with acceptable consequences. The proposed changes meet the criteria as specified in 10CFR50.67 and RG 1.183.

2.3 Elimination of Credit for the Control Room Envelope Pressurization System

The Control Room Envelope Pressurization System ensures that a positive pressure is maintained in the control room envelope for any event with the potential for radioactive releases. The positive pressure limits control room inleakage and consequently dose to the control room occupants so that regulatory dose limits are not exceeded. The proposed change and associated analysis does not credit the system in the calculation of the dose to the control room occupants resulting from the radiological event.

Since the system is not credited and the acceptance criterion is met for the radiological event, the elimination of the Technical Specification is proposed. Additionally, removal of the Technical Specification is further warranted since it does not meet the criteria in 10CFR50.36 for inclusion into the Technical Specifications.

2.4 Increase in the Acceptable Containment Leakage Rates

Containment leakage testing is performed to ensure that the leakage rate shall not exceed the leakage rate values as specified in the Technical Specifications. The Technical Specification acceptance criteria were developed to result in acceptable dose consequences following a radiological event. The proposed change in the acceptable leakage rate coupled with the analysis of the consequences from radiological events using the core and coolant source term specified by RG 1.183 continue to meet the acceptance criteria of 10CFR50.67 and RG 1.183.

2.5 Miscellaneous Bases Only Changes

Bases-only changes will be made primarily to change "10 CFR 100" to "10 CFR 50.67" or "Regulatory Guide 1.183", and to delete phrases such as "well within" and "small fraction of" that will not have regulatory significance with the AST design basis.

The Bases changes are provided for information only and will be implemented in accordance with 10CFR50.59.

2.6 Summary of Design and Licensing Basis Changes

This section provides a summary of the current design and licensing basis and the proposed changes. The summary is listed in Table 2.6-1. The existing analyses for the radiological events, as listed in Section 1.2, were performed at various times using different codes and/or hand calculations. The common element for these events is the assumption of the radiological source term documented in TID-14844 (Reference 7). The proposed amendment utilizes the approach in Regulatory Guide 1.183 and its supporting documents. Additionally, Westinghouse recently updated the thermal-hydraulic analyses supporting the SGTR, MSLB, LRA, and REA radiological calculations. This accounts for differences in some of the parameters listed in Table 2.6-1.

Table 2.6-1

Summary of Changes to the Design and Licensing Basis For the Radiological Event Analyses

Parameter	Current Basis	Proposed Basis
Alternate Source Term (Section 3.1)		
EAB Dose	First 2 hours of accident	Worst 2 hours of accident
Offsite Breathing Rates 0 – 8 hours 8 – 24 hours 24 – 720 hours	 3.47E-04 1.75E-04 2.32E-04	 3.50E-04 1.80E-04 2.30E-04
Dose Conversion Factors	ICRP30	FGR 11 and 12
RCS and Secondary Side Technical Specification Activity	FSAR Table 15.0-10	Table 3.3-1
Pre-accident Iodine Spike Activities	FSAR Table 15.0-12	Table 3.3-2
Coincident Spike	FSAR Table 15.0-13	Table 3.3-3 and 3.4-1

Table 2.6-1

**Summary of Changes to the Design and Licensing Basis
For the Radiological Event Analyses**

Parameter	Current Basis	Proposed Basis
Millstone Unit 3 Control Room (Section 3.1.5.4)		
Unfiltered Inleakage during Periods of Neutral Pressure	115 cfm	350 cfm
Unfiltered Inleakage during Periods of Positive Pressure	10 cfm	100 cfm
Intake Flow prior to Isolation	1450 cfm	1595 cfm
Time to isolate the Control Room	5 seconds	10 seconds
Control Room Envelope Pressurization System	Credited	Not credited (neutral pressure during operation)
X/Q's (unchanged for Millstone stack)	Murphy & Campe	ARCON96 (listed in Table 1.3-4)
Control Room Filter Efficiencies	Elemental, Aerosol and Organic: 95%	Elemental & Aerosol: 90% Organic: 70%
Loss-of Coolant Accident (Section 3.1)		
Containment Volume	2.32E+06 ft^3	2.35E+06 ft^3
Sprays	Quench spray only at 70.2 seconds	Quench Spray at 72.5 seconds Recirculation Spray at 14 minutes
Spray Coverage	Based on quench spray only	Increases when Recirculation Spray becomes effective (section 3.1.4.1)

Table 2.6-1

Summary of Changes to the Design and Licensing Basis For the Radiological Event Analyses

Parameter	Current Basis	Proposed Basis
Mixing Rate	Based on Quench Spray only	Decreases when Recirculation Spray becomes effective (section 3.1.4.1)
Particulate Iodine Removal Coefficient	Based on Quench Spray only	Increases when Recirculation Spray becomes effective (section 3.1.5)
Natural Deposition	50% of the Iodines plateout	"Powers" model used for aerosol in the unsprayed region
Containment Bypass Leak Rate	0.042 La	0.06 La
Iodine Chemical Form in Containment Atmosphere	5% Cesium Iodide 91% Elemental Iodine 4% Organic Iodine	95% Cesium Iodide 4.85% Elemental Iodine 0.15% Organic Iodine
Iodine Chemical Form in Containment Sump & RWST	5% Cesium Iodide 91% Elemental Iodine 4% Organic Iodine	97% Elemental Iodine 3% Organic Iodine
ECCS Leakage Initiation	220 seconds	640 seconds
RWST Backleakage	Leak rates and RWST airflow rate per Amendment 176	Section 3.1.5.3
Fuel Handling Accident (Section 3.2)		
Unfiltered Inleakage during Periods of Neutral Pressure	300 cfm	350 cfm
Unfiltered Inleakage during Periods of Positive Pressure	300 cfm	100 cfm

Table 2.6-1

Summary of Changes to the Design and Licensing Basis For the Radiological Event Analyses

Parameter	Current Basis	Proposed Basis
Steam Generator Tube Rupture Accident (Section 3.3)		
Steam Generator Mass Releases	FSAR Table 15.6.3-3	Table 3.3-4
Initial Steam Generator Mass	4.316E+07 grams	4.414E+07 grams
RCS Mass	2.359E+08 grams	2.358E+08 grams
Release Timing	FSAR Table 15.6.3-2	Table 3.3-4
Duration of Primary to Secondary Leakage for Intact Steam Generators	8 hours	18 hours
Main Steam Line Break Accident (Section 3.4)		
Steam Generator Mass Releases	FSAR Table 15.1-3	Table 3.4-2
Steam Generator Releases	167,000 lbm	164,200 lbm
Fuel Defects	0.29%	Technical Specification Limits on RCS Activity
Duration of Primary to Secondary Leakage for Affected Steam Generator	8 hours	55.2 hours
Duration of Primary to Secondary Leakage for Intact Steam Generators	8 hours	18 hours
Locked Rotor Accident (Section 3.5)		
Failed Fuel	6%	7%
Steam Generator Liquid Mass	103,000 lbm	97,222 lbm

Table 2.6-1

Summary of Changes to the Design and Licensing Basis For the Radiological Event Analyses

Parameter	Current Basis	Proposed Basis
RCCA Ejection Accident (Section 3.6)		
Containment Volume	2.32E+06 ft3	2.35E+06 ft3
Containment Bypass Leak Rate	0.042 La	0.06 La
Iodine Chemical Form in Containment Atmosphere	5% Cesium Iodide 91% Elemental Iodine 4% Organic Iodine	95% Cesium Iodide 4.85% Elemental Iodine 0.15% Organic Iodine
Iodine Chemical Form released from Steam Generator	5% Cesium Iodide 91% Elemental Iodine 4% Organic Iodine	97% Elemental Iodine 3% Organic Iodine
Steam Dump	40,604 lbm	200,000 lbm
Duration of Steam Dump	125 seconds	1,200 seconds
Duration of Steam Release for Cooldown	Not considered	16 hours (2 – 18 hours)
Time to initiate Safety Injection	1 minute	2 minutes
RCS Mass	520,000 lbm	5.194 E+05 lbm
Steam Generator Liquid Mass	103,000 lbm	97,222 lbm
Time for Primary System Pressure to fall below Secondary System Pressure	140 seconds	1,200 seconds

3.0 Radiological Event Re-analyses & Evaluation

As documented in Section 1.3.1, this application involves the reanalysis of the design basis radiological analyses for the following accidents:

- Loss-of-Coolant Accident (LOCA)
- Fuel Handling Accident (FHA)
- Steam Generator Tube Rupture (SGTR) Accident
- Main Steam Line Break (MSLB) accident
- Locked Rotor Accident (LRA)
- Rod Control Cluster Assembly (RCCA) Ejection Accident (REA).

These analyses have incorporated the features of the Alternate Source Term (AST), including the TEDE analysis methodology and modeling of plant systems and equipment operation that influence the events. The calculated radiological consequences are compared with the revised limits provided in 10 CFR 50.67(b)(2), and as clarified per the additional guidance in RG-1.183 for events with a higher probability of occurrence.

Dose calculations are performed at the Exclusion Area Boundary (EAB) for the worst 2-hour period, and for the Low Population Zone (LPZ) and Millstone Unit 3 Control Room for the duration of the accident (30 days). Dominion performed all the radiological consequence calculations for the AST with the RADTRAD-NAI and SCALE computer code systems (References 4 and 6) as discussed above. The dose acceptance criteria that apply for implementing the AST are provided in Table 3.0-1.

Table 3.0-1

Accident Dose Acceptance Criteria

Accident or Case	Control Room	EAB & LPZ
Design Basis LOCA	5 rem TEDE	25 rem TEDE
Steam Generator Tube Rupture		
Fuel Damage or Pre-accident Spike	5 rem TEDE	25 rem TEDE
Coincident Iodine Spike	5 rem TEDE	2.5 rem TEDE
Main Steam Line Break		
Fuel Damage or Pre-accident Spike	5 rem TEDE	25 rem TEDE
Coincident Iodine Spike	5 rem TEDE	2.5 rem TEDE
Locked Rotor Accident	5 rem TEDE	2.5 rem TEDE
RCCA Ejection Accident	5 rem TEDE	6.3 rem TEDE
Fuel Handling Accident	5 rem TEDE	6.3 rem TEDE

3.1 Design Basis Loss of Coolant Accident (LOCA) Reanalysis

This section describes the methods employed and results obtained from the LOCA design basis radiological analysis. The analysis includes dose from several sources. They are:

- Containment Leakage Plume,
- Emergency Core Cooling System (ECCS) Component Leakage
- Refueling Water Storage Tank Vent
- Shine from the plume,
- Shine from containment and
- Shine from the control room filter loading.

Doses are calculated at the Exclusion Area Boundary (EAB) for the worst-case two-hour period, at the Low Population Zone Boundary (LPZ), and in the Millstone Unit 3 Control Room. The methodology used to evaluate the Control Room and offsite doses resulting from a LOCA was consistent with RG 1.183 (Reference 1).

3.1.1 LOCA Scenario Description

The design basis LOCA scenario for radiological calculations is initiated assuming a major rupture of the primary reactor coolant system piping. In order to yield radioactive releases of the magnitude specified in RG 1.183, it is also assumed that the ECCS does not provide adequate core cooling, such that significant core melting occurs. This general scenario does not represent any specific accident sequence, but is representative of a class of severe damage incidents that were evaluated in the development of the RG 1.183 source term characteristics. Such a scenario would be expected to require multiple failures of systems and equipment and lies beyond the severity of incidents evaluated for design basis transient analysis.

3.1.2 LOCA Source Term Definition

RG 1.183 (Reference 1) provides explicit description of the key AST characteristics recommended for use in design basis radiological analyses. There are significant differences between the source term in RG 1.183 and the existing design basis source term documented in TID-14844 (Reference 7). The primary differences between the key characteristics of the two source terms are shown in Table 3.1-1 below.

Table 3.1-1

Comparison of TID-14844 and Regulatory Guide 1.183 Source Terms

Characteristic	TID Source Term	RG 1.183 Source Term
Core Fractions Released To Containment	Noble Gases 100% Iodine 50% (half of this plates out) Solids 1%	Noble Gases 100% Iodine 40% Cesium 30% Tellurium 5% Barium 2% Others – 0.02% to 0.25%
Timing of Release	Instantaneous	Released in Two Phases Over 1.8 hour Interval
Iodine Chemical and Physical Form	91% Inorganic Vapor 4% Organic Vapor 5% Aerosol	4.85% Inorganic Vapor 0.15% Organic Vapor 95% Aerosol
Solids	Ignored in Analysis	Treated as an Aerosol

RG-1.183 divides the releases from the core into two phases:

1) The Fuel Gap Release Phase during the first 30 minutes and
2) The Early In-vessel Release Phase in the subsequent 1.3 hours.

Table 3.1-2 shows the fractions of the total core inventory of various isotope groups that are assumed released in each of the two phases of the LOCA analysis.

Table 3.1-2

RG 1.183 Release Phases

	Core Release Fractions[a]	
Isotope Group	**Gap**	**Early In-Vessel**
Noble Gases[b]	0.05	0.95
Halogens	0.05	0.35
Alkali Metals	0.05	0.25
Tellurium	0	0.05
Barium, Strontium	0	0.02
Noble Metals	0	0.0025
Cerium	0	0.0005
Lanthanides	0	0.0002
Duration (hours)	0.5	1.3

a. Release duration apply only to the Containment release. The ECCS leakage portion of the analysis conservatively assumes that the entire core release fraction is in the containment sump from the start of the LOCA.

b. Noble Gases are not scrubbed from the containment atmosphere and therefore are not found in either the sump or ECCS fluid.

The core radionuclide inventory for use in determining source term releases was generated using the ORIGEN code. ORIGEN is part of the SCALE computer code system (Reference 6). Table 3.1-3 lists the 66 isotopes and the associated curies at the end of a fuel cycle that was input to RADTRAD-NAI. The core inventory used in the LOCA analysis is the identical source term that is used in the selective AST submittal for the Millstone Unit 3 FHA, which was approved by Reference 35. Table 3.1-3 also provides the CEDE and EDE dose conversion factors for each of the isotopes. These dose conversion factors were taken from Federal Guidance Reports 11 and 12 (References 8 and 9, respectively).

Table 3.1-3

Core Inventory and Dose Conversion Factors by Isotope

Isotope	Isotope Group	Curies	EDE $Sv\text{-}m^3/Bq\text{-}sec$	CEDE Sv/Bq
Kr-85	Noble gas	1.075E+06	1.190E-16	0.000E+00
Kr-85m	Noble gas	2.590E+07	7.480E-15	0.000E+00
Kr-87	Noble gas	4.755E+07	4.120E-14	0.000E+00
Kr-88	Noble gas	7.060E+07	1.020E-13	0.000E+00
Xe-133	Noble gas	1.980E+08	1.560E-15	0.000E+00
Xe-135	Noble gas	6.440E+07	1.190E-14	0.000E+00
Xe-135m	Noble gas	3.589E+07	2.040E-14	0.000E+00
Xe-138	Noble gas	8.610E+07	5.770E-14	0.000E+00
Br-84	Halogen	1.904E+07	9.410E-14	2.270E-11
I-131	Halogen	9.710E+07	1.820E-14	8.890E-09
I-132	Halogen	1.416E+08	1.120E-13	1.030E-10
I-133	Halogen	2.008E+08	2.940E-14	1.580E-09
I-134	Halogen	2.146E+08	1.300E-13	3.550E-11
I-135	Halogen	1.864E+08	7.980E-14	3.320E-10
Rb-86	Alkali Metal	2.170E+05	4.810E-15	1.790E-09
Rb-88	Alkali Metal	7.500E+07	3.360E-14	2.260E-11
Rb-89	Alkali Metal	6.400E+07	1.060E-13	1.160E-11
Cs-134	Alkali Metal	2.037E+07	7.570E-14	1.250E-08
Cs-136	Alkali Metal	6.270E+06	1.060E-13	1.980E-09
Cs-137	Alkali Metal	1.256E+07	7.740E-18	8.630E-09
Cs-138	Alkali Metal	1.711E+08	1.210E-13	2.740E-11
Sb-127	Tellurium	8.810E+06	3.330E-14	1.630E-09
Sb-129	Tellurium	3.080E+07	7.140E-14	1.740E-10
Te-127	Tellurium	8.700E+06	2.420E-16	8.600E-11
Te-127m	Tellurium	1.463E+06	1.470E-16	5.810E-09

Table 3.1-3

Core Inventory and Dose Conversion Factors by Isotope

Isotope	Isotope Group	Curies	EDE Sv-m^3/Bq-sec	CEDE Sv/Bq
Te-129	Tellurium	3.013E+07	2.750E-15	2.090E-11
Te-129m	Tellurium	6.140E+06	1.550E-15	6.470E-09
Te-131m	Tellurium	1.969E+07	7.010E-14	1.730E-09
Te-132	Tellurium	1.391E+08	1.030E-14	2.550E-09
Te-133m	Tellurium	7.620E+07	1.140E-13	1.170E-10
Te-134	Tellurium	1.438E+08	4.240E-14	3.440E-11
Sr-89	Barium-Strontium	1.056E+08	7.730E-17	1.120E-08
Sr-90	Barium-Strontium	9.330E+06	7.530E-18	3.510E-07
Sr-91	Barium-Strontium	1.276E+08	3.450E-14	4.490E-10
Sr-92	Barium-Strontium	1.278E+08	6.790E-14	2.180E-10
Ba-139	Barium-Strontium	1.722E+08	2.170E-15	4.640E-11
Ba-140	Barium-Strontium	1.800E+08	8.580E-15	1.010E-09
Mo-99	Noble Metal	1.826E+08	7.280E-15	1.070E-09
Rh-105	Noble Metal	1.052E+08	3.720E-15	2.580E-10
Ru-103	Noble Metal	1.606E+08	2.250E-14	2.420E-09
Ru-105	Noble Metal	1.137E+08	3.810E-14	1.230E-10
Ru-106	Noble Metal	6.120E+07	0.000E+00	1.290E-07
Tc-99m	Noble Metal	1.618E+08	5.890E-15	8.800E-12
Ce-141	Cerium	1.657E+08	3.430E-15	2.420E-09
Ce-143	Cerium	1.558E+08	1.290E-14	9.160E-10
Ce-144	Cerium	1.290E+08	8.530E-16	1.010E-07
Np-239	Cerium	2.080E+09	7.690E-15	6.780E-10

Table 3.1-3

Core Inventory and Dose Conversion Factors by Isotope

Isotope	Isotope Group	Curies	EDE $Sv\text{-}m^3/Bq\text{-}sec$	CEDE Sv/Bq
Pu-238	Cerium	4.083E+05	4.880E-18	7.790E-05
Pu-239	Cerium	3.404E+04	4.240E-18	8.330E-05
Pu-240	Cerium	4.810E+04	4.750E-18	8.330E-05
Pu-241	Cerium	1.511E+07	7.250E-20	1.340E-06
Am-241	Lanthanides	4.520E+03	8.180E-16	1.200E-04
Cm-242	Lanthanides	5.129E+06	5.690E-18	4.670E-06
Cm-244	Lanthanides	6.289E+05	4.910E-18	6.700E-05
La-140	Lanthanides	1.864E+08	1.170E-13	1.310E-09
La-141	Lanthanides	1.628E+08	2.390E-15	1.570E-10
La-142	Lanthanides	1.551E+08	1.440E-13	6.840E-11
Nb-95	Lanthanides	1.738E+08	3.740E-14	1.570E-09
Nd-147	Lanthanides	6.590E+07	6.190E-15	1.850E-09
Pr-143	Lanthanides	1.519E+08	2.100E-17	2.190E-09
Y-90	Lanthanides	9.700E+06	1.900E-16	2.280E-09
Y-91	Lanthanides	1.347E+08	2.600E-16	1.320E-08
Y-92	Lanthanides	1.366E+08	1.300E-14	2.110E-10
Y-93	Lanthanides	1.018E+08	4.800E-15	5.820E-10
Zr-95	Lanthanides	1.728E+08	3.600E-14	6.390E-09
Zr-97	Lanthanides	1.587E+08	9.020E-15	1.170E-09

3.1.3 Determination of Atmospheric Dispersion Factors (X/Q)

3.1.3.1 Millstone Unit 3 Control Room X/Q

The onsite atmospheric dispersion factors were calculated by Dominion using the ARCON96 code (Reference 17) and guidance from Regulatory Guide 1.194 (RG 1.194) (Reference 10). Enclosure 1 of this Attachment includes an electronic format of site meteorological data taken over the years 1997-2001 and the calculation of onsite X/Q values to the Millstone Unit 3 control room using ARCON96. The Control Room X/Qs were calculated for the LOCA for the following Millstone Unit 3 source points:

- Turbine Building Ventilation Vent
- Main Steam Valve Building (MSVB) Ventilation Exhaust
- Containment Enclosure Building
- Engineered Safeguards (ESF) Building Ventilation Exhaust
- Refueling Water Storage Tank (RWST) vent

The control room X/Q's from the Millstone stack were not recalculated using ARCON96. The values are consistent with current licensing basis and are based on Regulatory Guide 1.145 (Reference 36) methodology using fumigation conditions. These values are conservative when compared to the options recommended in RG 1.194 for determination of X/Q values from "Elevated (Stack) Releases."

The control room X/Q's used in the LOCA analysis are listed in Table 1.3-4.

3.1.3.2 Offsite (EAB & LPZ) X/Q

The Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) atmospheric dispersion factors are part of the existing design basis offsite dose calculations. The X/Q values, which were not revised for the AST analysis, are listed in Table 1.3-3. These offsite atmospheric dispersion factors were approved in Amendment No. 211, dated September 16, 2002 and November 25, 2002, (References 13 and 14) to Facility Operating License No. NPF-49 for Millstone Unit 3 regarding the revised Final Safety Analysis Report licensing basis for post-accident operation of the Supplementary Leakage Collection and Release System (TAC No. MB3700). The application was dated June 6, 1998 (Reference 19) and supplemented by References 20 through 28.

3.1.4 Determination of Containment Airborne Activity

3.1.4.1 Containment Sprays

The current licensing basis for the LOCA uses containment sprays to remove elemental and particulate iodine from the containment atmosphere. The use of containment sprays and approval of elemental and particulate iodine removal rates for quench spray were approved in Amendment No. 211, dated September 16, 2002 and November 25, 2002, (References 13 and 14) to Facility Operating License No. NPF-49 for Millstone Unit 3 regarding the revised Final Safety Analysis Report licensing basis for post-accident operation of the Supplementary Leakage Collection and Release System (TAC No. MB3700). The application was dated June 6, 1998 (Reference 19) and supplemented by References 20 through 28.

The percentage of containment that is covered by quench spray is 49.63%. The Quench Spray system becomes effective at 72.5 seconds. At 14 minutes post-LOCA the Recirculation Spray system becomes effective and the sprayed coverage of containment increases to 64.5% during the time when both spray systems are operating. The mixing rate during spray operation is 2 turnovers of the unsprayed volume per hour.

The elemental and particulate iodine removal rates due to sprays are listed in Table 3.1-6. These spray removal rates are used until the Quench Spray system is secured at 7,480 seconds. At that time further iodine removal is ignored due to sprays even though the Recirculation Spray system remains operating. An elemental iodine DF of 79 was calculated during the period that sprays are assumed operating. A particulate iodine DF of 49.5 was calculated up to 6,840 seconds, at which time it was reduced per Table 3.1-6.

3.4.1.2 Natural Deposition

A reduction in airborne radioactivity in the containment by natural deposition within containment was credited. The model used is described in NUREG/CR-6189 (Reference 12) and is incorporated into the RADTRAD computer code. This model is called the Powers model and it's used for aerosols in the unsprayed region and set for the 10th percentile.

3.1.5 LOCA Analysis Assumptions & Key Parameter Values

3.1.5.1 Method of Analysis

The RADTRAD-NAI code (Reference 3) is used to calculate the radiological consequences from airborne releases resulting from a LOCA at Millstone Unit 3 to the EAB, LPZ, and Millstone Unit 3 Control Room. The ORIGEN code is used to determine the grams of iodine in the core for calculating RWST backleakage. The QADS code was used to calculate the shine dose to the Control Room from containment shine and control room filter shine.

This analysis addresses a plant specific issue of unfiltered post-LOCA releases due to damper bypass and duct leakage from the plant ventilation system that was described and approved in Amendment No. 211, dated September 16, 2002 and November 25, 2002 (References 13 & 14), to Facility Operating License No. NPF-49 for Millstone Unit 3 regarding the revised Final Safety Analysis Report licensing basis for post-accident operation of the Supplementary Leakage Collection and Release System (TAC No. MB3700). The application was dated June 6, 1998 (Reference 19) and supplemented by References 20 through 28.

Amendment 211 identified potential release pathways from the 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. Amendment 211 also approved an operator action that would manually trip the breakers on selected fans at 1 hour and 20 minutes post-LOCA. This operator action is only credited in the control room habitability analysis. This licensing basis is further described in section 15.6.5.4, Radiological Consequences of a LOCA, in the Millstone Unit 3 FSAR. The AST analysis does not change the licensing basis for the post-accident operation of SLCRS as described and approved in Amendment 211.

3.1.5.2 Basic Data & Assumptions for LOCA

Table 3.1-4

Basic Data and Assumptions for LOCA

Parameter or Assumption / (Reference)	Value
Containment Leak Rate: (Technical Specifications)	0.3% by weight of the containment air per 24 hours (La)
Containment Bypass Leak Rate: (Technical Specifications & Conservative Assumption)	0.06La
Containment leak rate Reduction: (Reference 1)	50% after 24 hours (offsite) 50% after 1 hour (control room)
Secondary Containment Drawdown Time: (Technical Specifications)	2 minutes
Iodine Chemical Form in Containment Atmosphere: (Reference 1)	95% Cesium Iodide 4.85% Elemental Iodine 0.15% Organic Iodine
Iodine Chemical Form in the Sump and RWST: (Reference 1)	97% Elemental 3% Organic
Containment Sump pH:	at least 7
Dose Conversion Factors:	References 8 and 9
SLRCS Filter Efficiency: (Technical Specifications & Conservative Assumption)	95% all Iodines and Particulates

Table 3.1-4

Basic Data and Assumptions for LOCA

Parameter or Assumption / (Reference)	Value
Auxiliary Building Filter Efficiency: (Technical Specifications & Conservative Assumption)	95% all Iodines and Particulates
Quench Spray System Effective Period of Operation:	72.5 – 7,480 seconds
Recirculation Spray System Start Time: (Technical Specifications)	660 +/- 20 seconds
Recirculation Spray System Effectiveness Time:	14 minutes
Elemental Iodine Removal Coefficient:	20 per hour
Particulate Iodine Removal Coefficient for Quench Spray:	• DF < 50: 12.73 • DF $\geq$ 50: 1.27
Particulate Iodine Removal Coefficient for Quench and Recirculation Spray:	• DF < 50: 16.14 • DF $\geq$ 50: 1.61
Quench Spray Volume of Containment:	1,166,200 ft^3
Quench and Recirculation Spray Volume:	1,515,858 ft^3
ECCS System Leakage Outside Containment of Containment:	4,730 cc/hr
Minimum Available RWST Volume:	1,072,886 gallons
Minimum Quench Spray System Auto Trip Value:	47,652 gallons
RWST Maximum Fill Volume:	1,206,644 gallons

3.1.5.1 Containment Leakage Model

The containment leakage normally consists of filtered and bypass leakage. As stated in the data and assumptions, the total containment leak rate (La) is 0.3% per day. The bypass leak rate is assumed to be 0.06 * La or 0.018% per day after SLCRS drawdown time. The bypass leak rate bypasses the secondary containment and is released unfiltered at ground level directly from containment. The entire containment leak rate bypasses the secondary containment until the SLCRS drawdown time of 2 minutes (Reference 1). The leak rate is reduced by one-half (0.009% per day) at 24 hours for offsite calculations and at 1 hour for control room calculations.

The reduction in the containment leak rate by 50% at 1 hour for the control room analysis was approved in Amendment No. 211, dated September 16, 2002 and November 25, 2002 (References 13 and 14), to Facility Operating License No. NPF-49 for Millstone Unit 3 regarding the revised Final Safety Analysis Report licensing basis for post-accident operation of the Supplementary Leakage Collection and Release System (TAC No. MB3700). The application was dated June 6, 1998 (Reference 19) and supplemented by References 20 through 28. This reduction in containment leakage is based on the fact that the Millstone Unit 3 containment pressure is rapidly reduced compared to typical PWR's because of its original design as a negative pressure containment.

The collection, processing, and release of containment leakage vary depending on the location of the leak. Ventilation characteristics and release paths are different for each building comprising the secondary containment. Tables 15.6-9 and 15.6-12 of the Millstone Unit 3 FSAR describe the ventilation characteristics and release paths.

3.1.5.2 Model of ECCS Leakage

The Emergency Core Cooling System (ECCS) fluid consists of the contaminated water in the sump of the containment. This water contains 40% of the core inventory of iodine, 5% released to the sump water during the gap release phase (30 minutes) and 35% released to the sump water during the early in-vessel phase during the next 1.3 hours. During a LOCA the highly radioactive ECCS fluid is pumped from the containment sump to the recirculation spray headers and sprayed back into the containment sump. Also, following a design basis LOCA, valve realignment occurs to switch the suction water source for the ECCS from RWST to the containment sump.

ECCS leakage develops when ESF systems circulate sump water outside containment and leaks develop through packing glands, pump shaft seals and flanged connections. The Technical Specification 6.8.4a, Primary Coolant Sources Outside Containment Program Manual calculates this leakage at 4,780 cc/hr. In accordance with Reference 1, the ECCS analysis makes use of 10,000 cc/hr for ECCS leakage. The leakage of recirculating sump fluids commences at 640 seconds, which is the earliest time of recirculation.

The temperature of the containment sump is conservatively assumed to reach a maximum of 240 degrees F. At this maximum temperature, a flash fraction of 0.03 is calculated. However, per the guidance of RG-1.183, a conservative flash fraction of 0.1 was used for the ECCS leakage during the entire event. The water volume of the sump at 640 seconds is 1.068E+05 gallons and increases to 4.074E+05 gallons at 0.8217 hours, where it is assumed to remain constant.

3.1.5.3 Model of ECCS Back Leakage to Refueling Water Storage Tank

Following a design basis LOCA, valve realignment occurs to switch the suction water source for the ECCS from the Refueling Water Storage Tank (RWST) to the containment sump. In this configuration, Motor Operated Valves (MOVs) and check valves in the normal suction line from the RWST and MOVs in the recirculation line provide isolation between this contaminated flow stream and the RWST. RADTRAD-NAI is used to model leakage of ECCS fluid through these valves back into the RWST with subsequent leakage of the evolved iodine through the vent at the top of the RWST to the environment.

The RADTRAD-NAI source term used to model the ECCS leakage into the RWST contains only the iodine isotopes. Forty percent of the core inventory of iodine isotopes was modeled as being instantaneously transported from the core to the containment sump. The iodine form is 97% elemental and 3% organic in accordance with RG-1.183.

New times, new flow rates and new contaminated volumes discharged to the RWST from assuming higher leak rates in the RWST backleakage paths have been calculated. The leak paths back to the RWST are:

- CHS Alternate Recirculation Leakage
- RHR Leakage through V*43
- SIH Pump Recirculation
- RHRS A and B suction
- CHS Suction
- SIH Suction

The leakage paths and methodology to calculate times, flow rates and volumes were approved in Amendment 176 (Reference 29) to Facility Operating License No. NPF-49 for Millstone Unit 3, in response to the application dated May 7, 1198, (Reference 30) as supplemented January 22, 1999 (Reference 31) regarding RWST backleakage (TAC No. MA1749).

Table 3.1-5 summarizes the results of the above leakages for the 7 sources of backleakage to the RWST.

Table 3.1-5

Contaminated Inflow to RWST

Source	Time, hours	Flow Rate, gpm
CHS Suction	137.66	0.20
RHRS A Suction	126.25	0.20
RHRS B Suction	144.43	0.20
SIH Recirculation	8.50	0.20
RHRS Recirculation	29.83	0.60
CHS Recirculation	36.93	0.20
SIH Suction	67.48	0.20

Using the methodology approved in Amendment 176, the time for contaminated sump water to reach the RWST is based on the calculated flow rates and the volume of clean water in the associated piping. The time required to displace the clean volume is reduced by 50% to account for mixing in the lines. This is considered a reasonable assumption since the sump fluid is relatively cool and thermal mixing will be minimal. In addition, the lines are isolated and stagnant except for minor leakage rates and the mixing due to flow is negligible. Table 3.1-6 reduces the times in Table 3.1-5 by 50%, integrates the flow rates over time and calculates the total contaminated volume discharged to the RWST over the 30 day LOCA period.

Table 3.1-6

Summary of Times, Integrated Flow Rates & Volumes for RWST Backleakage

Time (hrs)	Flow Rate (cfm)	Volume (ft3)	
4.25	0.03	0.00	SIS R
14.91	0.11	17.11	RHS R
18.46	0.13	39.88	CHS R
33.74	0.16	162.41	SIH S
63.13	0.19	445.21	RHRS A
68.83	0.21	509.27	CHS S
72.21	0.24	552.68	RHR B
720.0	0.24	9,904.49	@ 30 days

For the analysis of the partition coefficient, the amount of water remaining in the RWST at the end of the injection phase is conservatively taken to correspond to the lowest possible value; the minimum QSS auto trip value or 47,652 gallons. The RWST airflow rate of 8.7 cfm was determined by making use of the ideal gas law and expected volumetric change. The latter was based on a conservative rise in air temperature within the RWST as a result of solar heating. The air released from the RWST will be free of radioactivity until the backleakage reaches the RWST at 4.25 hours post-LOCA.

The partition coefficient (PC) applicable to the iodines in the RWST water is based upon information in Reference 15. For this application, the RWST was assumed to behave like a closed system for the establishment of equilibrium conditions between the water and air. ORIGEN was used to calculate the quantity of grams of iodine in the core at 20,000 grams. The fraction of iodine released during the LOCA is 0.4, resulting in the grams of iodine in the sump at 8.0E+03. The volume of liquid in the sump is the sum of 1,160,776 gallons, resulting in an iodine concentration in the sump of 7 mgrams / gallon. Total volume transferred to the RWST over the 30 days as a result of backleakage is 7.41E+04 gallons resulting in a total of 5.187E+05 mgrams of iodine transferred to the RWST. The maximum concentration of iodines in the RWST is 4.3 mgrams/gallon or 1.2 mgrams/liter. The PC of 4,000 corresponds to an iodine concentration of 1.2 mgrams/liter, taken from Reference 15. A PC of 4,000 results in a DF of 450. A DF of 100 was used for conservatism.

3.1.5.4 Millstone Unit 3 Control Room

The control room volume is 2.38E5 ft^3. The LOCA causes a Control Building Isolation (CBI) signal to isolate the control room (current Technical Specifications). The control building is isolated within 5 seconds after a CBI signal. According to Reference 1, the onset of the gap release does not start until 30 seconds post-LOCA. Therefore the control room will be isolated prior to the arrival of the radioactive release.

The following is taken from the current Technical Specifications.

"After a 60 second time delay the control room envelope pressurizes to greater than or equal to 1/8-inch water gauge relative to adjacent areas and the outside atmosphere. The positive pressure is maintained for greater than or equal to 60 minutes."

The timing and operation of the Control Room Envelope Pressurization System and Control Room Emergency Ventilation System is described in Section 9.4 of the Millstone Unit 3 FSAR. The timing and operation of these systems were approved in Amendment No. 211, dated September 16, 2002 and November 25, 2002, to Facility Operating License No. NPF-49 for Millstone Unit 3 regarding the revised Final Safety Analysis Report licensing basis for post-accident operation of the Supplementary Leakage Collection and Release System (TAC No. MB3700). The application was dated June 6, 1998 and 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.

In the LOCA analyses the Control Room Envelope Pressurization System is not credited with operating and providing a positive pressure in the control room. Therefore, during the one-hour period that the Control Room Envelope Pressurization System should be operating, the control room is assumed to be at a neutral pressure. During periods of neutral pressure in the Millstone Unit 3 control room, unfiltered inleakage is assumed to be at the analysis limit of 350 cfm. During periods of positive pressure in the Millstone Unit 3 control room unfiltered inleakage is assumed to be at the analysis limit of 100 cfm.

The Control Room Emergency Ventilation System filter efficiencies are conservatively assumed at 90% for both elemental and aerosol and 70% for organic iodines.

The post LOCA dose consequences to the Millstone Unit 3 control room are due to the following sources:

1. 1. airborne contribution
   - containment leakage
   - ESF leakage
   - RWST backflow
2. external sources
   - control room filter shine
   - cloud shine
   - RWST direct shine
   - containment direct shine

The doses due to external sources were calculated using data from Tables 1.3-1, 1.3-2, and Section 3.1.5

3.1.6 LOCA Results

Table 3.1-7 lists TEDE to the EAB and LPZ from a LOCA at Millstone Unit 3. The dose to the EAB and LPZ is less than the 25 rem TEDE limit stated in 10CFR50.67 and Regulatory Guide 1.183. The EAB dose represents the worst 2-hour dose for each release pathway.

The dose to the Millstone Unit 3 control room is less than the 5 rem TEDE limit specified in in 10CFR50.67 and Regulatory Guide 1.183.

Table 3.1-7

TEDE from a Millstone Unit 3 LOCA

Location	TEDE (rem)
EAB	9.1E+00
LPZ	4.5E+00
Millstone Unit 3 Control Room	3.4E+00

3.2 Fuel Handling Accident (FHA)

This section describes the methods employed and results of the Fuel Handling Accident (FHA) design basis radiological analysis. The analysis includes doses associated with release of gap activity from a fuel assembly either inside containment or in the Fuel Building. Doses were calculated at the Exclusion Area Boundary (EAB), at the Low Population Zone (LPZ) boundary, and in the Unit 3 control room. The methodology used to evaluate the control room and offsite doses resulting from the FHA is consistent with RG 1.183 in conjunction with TEDE radiological units and limits, ARCON96 based onsite atmospheric dispersion factors, and Federal Guidance Reports No. 11 and 12 dose conversion factors.

The FHA recently approved in Amendment 219 (Reference 35) for selective implementation of the AST differs from the FHA in this amendment request by the following:

1) Revised Control Room X/Qs (now based on ARCON96),
2) Changes were made to Control Room Inleakage assumptions (larger inleakage rates),
3) Reduced Control Room filtration efficiency (from 95% to 90/90/70% particulate/ elemental/ organic) and
4) No credit for the Control Room Envelope Pressurization System.

3.2.1 FHA Scenario Description

The design basis scenario for the radiological analysis of the FHA assumes that cladding damage has occurred to all of the fuel rods in one fuel assembly plus 50 rods in the struck assembly. This scenario is unchanged from the assumption in the existing UFSAR analysis. The rods are assumed to instantaneously release their fission gas contents to the water surrounding the fuel assemblies. The analyses include the evaluation of FHA cases that occur in both the containment and the Fuel Building. Essentially all radioactivity released from the damaged fuel is assumed to release over a two hour period through an open penetration in the containment or the Fuel Building.

3.2.2 FHA Source Term Definition

In accordance with Regulatory Position 3 of RG 1.183 the core source was determined using ORIGEN to evaluate multiple cycle designs (based on the Dominion fuel management scheme for enrichment and burnup). This core inventory was used and approved in the FHA selective implementation of the AST in Amendment 219 and is described in the LOCA scenario (Section 3.1.2) and is used for the FHA with a 100-hour decay time.

For the FHA analyses, the core inventory was used to calculate the gap activity of one fuel assembly plus 50 rods for input to RADTRAD-NAI. The amount of fuel damage is the same whether the FHA is in the Fuel Building or Containment. Therefore, the only variable between FHA in the Containment or in the Fuel Building is the release point. Consistent with Amendment 219 for the selective implementation of the AST, the FHA analyses in this amendment request assume the resulting chemical form of the radioiodine in the water is 99.85% elemental iodine and 0.15% organic iodide.

3.2.3 FHA Release Transport

This evaluation does not credit operability or operation of the Containment purge system, Auxiliary Building or Fuel Building ventilation. This evaluation assumes that the personnel hatch, equipment hatch and penetrations are open for the duration of the 2 hour release.

Releases from the Fuel Building or Containment to the environment are at a rate of 3.454 air changes per hour. This assures that greater than 99.9% of the activity in the Fuel Building and Containment analyses were released within 2 hours. The release rate is conservative in that it biases the bulk of the release (i.e., > 80%) to occur within the first half hour of the event. No credit is taken for filtration of the release from either the Fuel Building or Containment. Additionally, no credit is taken for dilution or mixing of the activity released to the Fuel Building or Containment air volumes.

If the Containment is not isolated, the release from a FHA inside the Containment could be from:

1) The equipment hatch or penetrations via the Enclosure Building,
2) The personnel hatch or penetrations to the auxiliary building and discharged out the Turbine Building Ventilation Stack,
3) Leakage discharged through purge to the Turbine Building Ventilation Stack.

For a FHA in the Fuel Building, the release is from either the Turbine Building Ventilation Stack or the Fuel Building roll-up doors.

The most conservative release point to the Control Room is an unfiltered release from the Turbine Building Ventilation Stack. For conservatism in calculating off-site doses, the release is discharged unfiltered out the Equipment Hatch via the Containment Enclosure Building. This is because the EAB Containment Enclosure Building X/Q is higher than for the Ventilation Stack release. This modeling is consistent with Amendment 219, approved, March 17, 2004.

3.2.4 Determination of Atmospheric Dispersion Factors (X/Q)

3.2.4.1 Control Room Atmospheric Dispersion Factors

The onsite atmospheric dispersion factors were calculated by Dominion using the ARCON96 code and guidance from RG-1.194 [Reference 10] for the Control Rooms. Site meteorological data taken over the years 1997-2001 were used in the evaluations. Control room X/Q values were calculated at the Turbine Building Ventilation Stack, and Containment Enclosure Building. The control room atmospheric dispersion factors are presented in Table 1.3 4.

3.2.4.2 Offsite Atmospheric Dispersion Factors (X/Q)

The offsite atmospheric dispersion factors (EAB and LPZ) used for the FHA analysis are the same as those used for LOCA. They are reported in Table 1.3-3.

3.2.5 FHA Analysis Assumptions & Key Parameter Values

The basic data and assumptions are listed below in Table 3.2-1.

Table 3.2-1

Data and Assumptions for the Fuel Handling Accident Analysis

Data / Assumption	Value
Gap Fractions	Noble Gases: 10% Halogens: 8%
Pool Decontamination Factor:	Noble Gases: 1 Iodines: 200 (effective DF)
Release Point:	• Turbine Building Ventilation Stack or • Enclosure Building / Containment Ground
Decay Time:	100 hours
Radial Peaking Factor:	1.7
Duration of Release to the Environment:	2 hours

Table 3.2-1

Data and Assumptions for the Fuel Handling Accident Analysis

Fuel Damage:	1 assembly plus 50 rods out of a total of 193 assemblies in a core
Plume and Filter Shine are accounted in the Control Room doses.	
Control Room Ventilation Timing • T= 0 seconds Accident Initiation Unfiltered Intake Flow –1595 cfm • T= 5 seconds Control Building Isolation (CBI) Signal generated Control Room Envelope Pressurization System receives CBI signal • T= 10 seconds Control Room Isolates on Radiation Monitor Signal Intake Flow – 0 cfm Unfiltered Inleakage Flow –350 cfm • T= 1 minute, 5 seconds Assumes 1 minute delay for Control Room Envelope Pressurization System Bottles response time No credit is taken for operability of Control Room Envelope Pressurization System Unfiltered Inleakage Flow –350 cfm • T= 1 hour, 41 minutes, 5 seconds (= 1.685 hours) Assume 40 minute delay for operator action Control Room is pressurized Filtered Intake Flow – 230 cfm Unfiltered Inleakage Flow – 100 cfm Filtered Recirculation Flow – 666 cfm • T= 720 hours • Filtered Recirculation Flow – 666 cfm	

3.2.6 FHA Analysis Results

The offsite and control room doses are listed below. The Millstone Unit 3 Fuel Handling Accident assumes a two-hour release without building integrity for the Containment FHA and without Fuel Building Filtration for the Spent Fuel Pool FHA. The associated worst case TEDE is presented in Table 3.2-2. All doses are less than the limits specified in Regulatory Guide 1.183 and 10 CFR 50.67.

Table 3.2-2

Dose Summary for the Fuel Handling Accident Analysis

Location	TEDE (rem)	Limits (rem)
EAB	2.4E+00	6.3
LPZ	1.3E-01	6.3
Millstone Unit 3 Control Room	4.9E+00	5

3.3 Steam Generator Tube Rupture Accident

This section describes the methods employed and the results of the Steam Generator Tube Rupture (SGTR) design basis radiological analysis. This analysis included doses associated with the releases of the radioactive material initially present in primary liquid, secondary liquid and iodine spiking. Doses were calculated at the Exclusion Area Boundary (EAB), at the Low Population Zone (LPZ), and in the control room. The methodology used to evaluate the control room and offsite doses resulting from the SGTR accident was consistent with Regulatory Guide 1.183 in conjunction with TEDE radiological units and limits, used ARCON96 based onsite atmospheric dispersion factors, and utilized Federal Guidance Reports (FGR) No. 11 and 12 dose conversion factors.

3.3.1 SGTR Scenario Description

A SGTR is a break in a tube carrying primary coolant through the steam generator. This postulated break allows primary liquid to leak to the secondary side of one of the steam generators (denoted as the affected generator) with an assumed release to the environment through the steam generator Atmospheric Dump Valves (ADVs). The ADV on the affected steam generator is assumed to open to control steam generator pressure at the beginning of the event, and then fail fully open after operator action was taken to close the steam generator ADV. The affected generator discharges steam to the environment for 2946 seconds (0.8183 hours) until the generator is isolated a second time by closure of the steam generator Atmospheric Dump Block Valve (ADBV). Break flow into the affected steam generator continues until 5596 seconds (1.554 hours), at which time the RCS is at a lower pressure. Additional releases from the affected steam generator are modeled from 2-8 hours to complete depressurization of the steam generator early in the event to maximize the dose consequences. Depressurization of the steam generator is necessary to initiate Residual Heat Removal System (RHRS) cooling.

The intact generator (3 generators modeled as one) discharges steam for a period of 18 hours until the primary system has cooled sufficiently to allow a switchover to the RHRS, at 11 hours, plus a 7 hour period of concurrent steaming. The additional 7 hours of steaming are required to reduce the system heatload to the point where RHRS can remove all the decay heat crediting only safety grade equipment to achieve cold shutdown and steaming is no longer required for cooldown. No fuel damage is predicted as a result of a SGTR. Therefore, consistent with the current licensing analysis basis, the SGTR analysis was performed assuming both a pre-accident iodine spike and a concurrent accident iodine spike. In addition, the impact of a coincident loss-of-offsite power (LOOP) at the time of tube rupture was considered.

In accordance with Regulatory Guide 1.183, The release of noble gases has been analyzed. Without credit for holdup in that scenario, the affected generator discharges steam to the environment for 1.554 hours after which the break flow stops and the generator block valve is closed. An alternate scenario for the affected steam generator was also evaluated for dose consequences associated with noble gases. Holdup of noble gases in the affected steam generator has been credited in the alternate scenario because of operator action to close the ADBV at 0.8183 hours with the break flow continuing to enter the generator until 1.554 hours and subsequent release at 2 hours. From the period of 0.8183 hours to 2 hours, noble gases are held up in the generator.

3.3.2 SGTR Source Term Definition

Initial radionuclide concentrations in the primary and secondary systems for the SGTR accident are determined based on the maximum Technical Specification levels of activity. The SGTR accident analysis indicates that no fuel rod failures occur as a result of these transients. Thus, radioactive material releases were determined by the radionuclide concentrations initially present in primary liquid, secondary liquid, and iodine spiking. These values are the starting point for determining the curie input for the RADTRAD-NAI code runs.

Regulatory Guide 1.183 specifies that the released activities should be the maximum allowed by the Technical Specifications. Table 3.3-1 lists all the primary and secondary liquid radionuclide concentrations that are used in the analysis. Primary side concentrations are based on the Technical Specification limits of 100/ E_{bar} for gross gamma and 1.0 uCi/gm Dose Equivalent (DEQ) I-131 for iodines. Secondary side concentrations are based on the Technical Specification limit of 0.1 uCi/gm DEQ I-131 for iodine. In addition, since there is not a Technical Specification limit for the secondary side gross gamma activity, one was derived from the design basis steam generator liquid activity to ensure that a suitably conservative source term was used.

Regulatory Guide 1.183 stipulates that SGTR accidents consider iodine spiking above the value allowed for normal operations based both on a pre-accident iodine spike and a concurrent accident spike. For Millstone Unit 3, the maximum iodine concentration allowed by Technical Specifications as the result of an iodine spike is 60 uCi/gm dose equivalent I-131. This value is treated as the pre-accident iodine spike and is listed in Table 3.3-2. Regulatory Guide 1.183 defines a concurrent iodine spike as an accident initiated value 335 times the appearance rate corresponding to the Technical Specification limit for normal operation (1 uCi/gm DEQ I-131 RCS TS limit) for a period of 8 hours. The concurrent iodine spike appearance rates based on 335 times the 1.0 uCi/gm DEQ I-131 concentration are listed in Table 3.3-3. Appearance rates address the issues raised by NSAL-00-004 [Reference 32].

The dose conversion factors used to calculate the TEDE doses and DEQ I-131 for the Steam Generator Tube Rupture accident were taken from Table 3.1-3 for the isotopes required by Regulatory Guide 1.183 for the SGTR analysis.

Table 3.3-1

Primary Coolant and Secondary Side Liquid Nuclide Concentrations

Nuclides	RCS, uCi/gm	Secondary Side Water, uCi/gm
I131	7.721E-01	7.956E-02
I132	2.716E-01	2.650E-02
I133	1.245E+00	1.132E-01
I134	1.700E-01	4.804E-03
I135	6.531E-01	4.963E-02
Kr85m	4.654E+00	0.000E+00
Kr85	9.324E-02	0.000E+00
Kr87	3.340E+00	0.000E+00
Kr88	9.137E+00	0.000E+00
Xe133	7.112E+01	0.000E+00
Xe135m	3.118E+00	0.000E+00
Xe135	1.386E+01	0.000E+00
Xe138	1.618E+00	0.000E+00
Rb86	7.301E-04	8.437E-07
Rb88	9.137E+00	5.138E-04
Rb89	2.796E-01	1.358E-05
Cs134	8.927E-01	1.877E-03
Cs136	4.600E-01	9.384E-04
Cs137	4.497E+00	9.411E-03
Cs138	2.354E+00	2.401E-04
Co58	4.597E-02	5.408E-05
Co60	5.679E-03	6.760E-06
Br84	1.115E-01	9.762E-06
Sr89	1.169E-02	1.366E-05
Sr90	4.624E-04	5.408E-07
Sr91	5.398E-03	3.921E-06
Sr92	2.081E-03	7.653E-07
Y90	5.627E-04	1.155E-06
Y91	1.875E-03	3.948E-06
Y92	2.024E-03	1.739E-06
Y93	9.313E-04	9.627E-07

Nuclides	RCS, uCi/gm	Secondary Side Water, uCi/gm
Zr95	1.929E-03	2.247E-06
Zr97	1.127E-03	9.762E-07
Nb95	2.005E-03	2.342E-06
Mo99	9.240E+00	1.679E-02
Tc99m	5.211E+00	1.239E-02
Ru103	9.302E-04	1.082E-06
Ru105	8.248E-05	4.164E-08
Ru106	8.840E-05	1.033E-07
Rh105	2.362E-04	2.374E-07
Sb127	3.905E-05	4.300E-08
Sb129	9.021E-05	4.489E-08
Te127m	5.706E-03	6.652E-06
Te127	2.880E-03	4.651E-06
Te129m	1.059E-01	1.230E-04
Te129	5.933E-02	1.141E-04
Te131m	6.349E-02	6.220E-05
Te132	7.334E-01	7.977E-04
Te133m	6.998E-02	1.074E-05
Te134	8.451E-02	1.079E-05
Ba139	2.096E-01	5.354E-05
Ba140	1.193E-02	1.368E-05
La140	4.086E-03	5.895E-06
La141	7.499E-04	3.597E-07
La142	2.282E-06	5.571E-10
Ce141	1.893E-03	2.199E-06
Ce143	1.408E-03	1.398E-06
Ce144	1.343E-03	1.568E-06
Pr143	1.834E-03	2.131E-06
Nd147	6.593E-04	7.545E-07
Np239	1.055E-02	1.117E-05

Table 3.3-2

Pre-accident Iodine Spike RCS Concentration

Nuclide	Iodine Activity in RCS at 1.0 DEQ I-131 uCi/gm	Iodine Activity in RCS at 60 times 1.0 DEQ I-131 uCi/gm
I131	7.72E-01	4.63E+01
I132	2.72E-01	1.63E+01
I133	1.25E+00	7.47E+01
I134	1.70E-01	1.02E+01
I135	6.53E-01	3.92E+01

Table 3.3-3

Concurrent Iodine Spike RCS Concentration

Nuclide	Appearance rate for 1 uCi/gm DEQ I-131, uCi/sec	Spike = 335, SGTR Appearance Rate, uCi/sec
I131	5.24E+03	1.75E+06
I132	7.13E+03	2.39E+06
I133	1.09E+04	3.64E+06
I134	9.91E+03	3.32E+06
I135	8.77E+03	2.94E+06

3.3.3 Release Transport

Affected Steam Generators

The source term resulting from the radionuclides in the primary system coolant and from the iodine spiking in the primary system is transported to the affected steam generator by the break flow. The break flow is terminated after 1.554 hours. A fraction of the break flow is assumed to flash to steam in the affected generator and to pass directly into the steam space of the affected generator with no credit taken for scrubbing by the steam generator liquid. The radionuclides entering the steam space as the result of flashing pass directly to the environment through the Steam Generator ADVs. The remainder of the break flow enters the steam generator liquid. Releases of radionuclides initially in the steam generator liquid and those entering the steam generator from the break flow are released as a result of secondary liquid boiling including an allowance for a partition factor of 100 for all non-noble gas isotopes. Thus 1% of the iodines and particulates are released from the steam generator liquid to the environment along with the steam flow (moisture carryover is not actually modeled but is instead bounded by application of the partitioning factor). All noble gases are released from the primary system to the environment without reduction or mitigation. As was mentioned previously, an alternate noble gas release scenario was evaluated which considered isolation of the affected steam generator release to the environment by operator action at 0.8183 hours after the tube rupture had occurred, while the break flow continues into the generator until 1.554 hours, and subsequent allowance for depressurization of the generator and release of the accumulated contents from 2 to 8 hours post-accident. The transport model utilized for iodine and particulates was consistent with Appendix E of Regulatory Guide 1.183.

Intact Steam Generators

The source term resulting from the radionuclides in the primary system coolant and from the iodine spiking in the primary system is transported to the intact generators by the leak rate limiting condition for operation (1gpm) specified in the Technical Specifications. All radionuclides in the primary coolant leaking into the intact generator are assumed to enter the steam generator liquid. Releases of radionuclides initially in the steam generator liquid and those entering the steam generator from the leakage flow are released as a result of secondary liquid boiling, including an allowance for a partition factor of 100 for all non-noble gas isotopes. Thus 1% of the iodines and particulates are assumed to pass into the steam space and then directly to the environment. All noble gases that are released from the primary system to the intact generator are released to the environment without reduction or mitigation. Releases were assumed to continue from the intact generator for a period of 18 hours until the primary system cools to below 350°F and the RHRS can remove 100% of decay heat with no requirement for steaming to augment cooldown. The 18-hour steaming period is based on the time necessary to cooldown crediting safety grade equipment only.

3.3.4 Determination of Atmospheric Dispersion Factors

3.3.4.1 Control Room Atmospheric Dispersion Factors

The onsite atmospheric dispersion factors were calculated by Dominion using the ARCON96 code and guidance from RG-1.194 [Reference 10] for the control room. Site meteorological data taken over the years 1997-2001 were used in the evaluations. The control room atmospheric dispersion factors are presented in Table 1.3-4 under "Main Steam Valve Building Ventilation Exhaust".

3.3.4.2 Offsite Atmospheric Dispersion Factors

The EAB and LPZ X/Q values used in the SGTR analysis are listed in Table 1.3-3. The EAB values are listed under "Other Release Points" and the LPZ values are listed under "Non-Millstone Stack Release Points".

3.3.5 SGTR Key Analysis Assumptions and Inputs

The Basic Data and Assumptions are listed below in Table 3.3-4. A time line of events is provided in Table 3.3.5. Steam and break flow data are listed in Tables 3.3-6 to 3.3-8. Generic data such as control room information is available in Table 1.3-1.

Table 3.3-4

Basic Data and Assumptions for the SGTR Accident

Data / Assumption	Value
Primary to Secondary Leak Rate – Technical	1 gpm (intact steam generator)

Table 3.3-4

Basic Data and Assumptions for the SGTR Accident

Specification limit	
Release coincident with loss of off-site power	

Table 3.3-4

Basic Data and Assumptions for the SGTR Accident

Data / Assumption	Value
Release points: (On Loss Of Off-site Power, the condenser is not available for cooling, an ADV on the affected steam generator is assumed stuck open until closed by an operator. Additional cooling is by ADVs on intact steam generators.)	Steam Generator Atmospheric Dump Valves (ADVs)
Credited Operator Actions • Affected Steam Generator Close stuck open ADV after 1742 seconds (29 minutes) Close failed open ADV after an additional 1200 seconds (20 minutes)	0.8183 hrs.
Credited Operator Actions • Intact Steam Generators Actions commensurate with cooldown using only safety grade equipment	18 hrs.
Iodine chemical form (%) released from steam generators to environment	Elemental 97 Organic 3
Iodine Partitioning	PC for iodine = 100
Moisture Carryover in Intact Steam Generators	1%
Tube Uncovery.	• No tube bundle uncovery assumed. • Assumption consistent with conclusions in WCAP-13132 (Reference 33)
Release to Environment Duration Intact steam generators Affected steam generator	 0 – 18 hours 0 – 0.8183 hours & 2 – 8 hours

Table 3.3-4

Basic Data and Assumptions for the SGTR Accident

Release to Environment Duration Affected steam generator (alternate scenario – noble gas only)	From 0 to 1.554 hours (break flow stops after 1.554 hours)
Data / Assumption	**Value**
Initial Steam Generator Steam Mass	8,870 lbm / steam generator
Initial Steam Generator Liquid Mass	97,222 lbm / steam generator (updated from Westinghouse thermal-hydraulic analysis)
Control Room Ventilation Timing	Same as for the Fuel Handling Accident (Table 3.2.1)
Assumption: Dose consequences from release of initial secondary side steam is not significant	
Control Room Plume and Filter Shine Dose to the Millstone Unit 3 Control Room • Conservatively set the same as the LOCA analyses	

Table 3.3-5

Time Line of Events

Time, post accident		Event
seconds	hours	
0	0	SGTR – ADV sticks open LOOP
10	0.0028	Control Room Isolates
109	0.03028	Reactor Trip
381	0.1058	SI Actuated
1742	0.4839	Affected SG Isolated
1744	0.4844	Affected SG ADV Fails Open
2946	0.8183	Affected SGADBV Closed (Release Terminated)
5596	1.554	Break Flow Terminated
6065	1.685	Control Room Emergency Ventilation
7200	2	Affected SG Depressurization Initiated (Release Re-Initiated)
	8	Affected SG Depressurized, (Release Terminated)
	11	RCS at 350°F; RHRS Placed In Service
	18	RHRS capable of 100% of cooldown, Intact SG release stops
	55	RCS at 212°F Primary-to-Secondary Leak Stops
	720	Event Terminated

Table 3.3-6

RCS Break Flow to Affected Steam Generator

Time period		Total Break Flow Rate	Flashed Break Flow Rate	Liquid Break Flow Rate
From	To			
(hour)		lbm/min	lbm/min	lbm/min
0	0.03028	2642	423	2219
0.03028	0.03153	2390	423	1967
0.03153	0.4063	2390	149	2241
0.4063	0.7357	2390	377	2013
0.7357	0.8890	2390	161	2229
0.8890	0.9279	2390	41	2349
0.9279	1.554	2390	0	2390
1.554	2	0	0	0

Table 3.3-7

Affected Steam Generator Steam Release to Environment

Time period, sec		Time period, hour		Release Rate,
From	To	From	To	(lbm/min)
0	109	0	0.03028	64569
109	2946	0.03028	0.8183	3923
2946	7200	0.8183	2	0
7200	28800	2	8	95

Table 3.3-8

Intact Steam Generator Steam Release to the Environment

Time period, sec		Time period, hour		Release Rate (lbm/min)
From	To	From	To	
0	109	0	0.03028	192220
109	1762	0.03028	0.4895	4540
1762	5596	0.4895	1.554	3641
5596	7200	1.554	2	4970
7200	28800	2	8	2614
28800	39600	8	11	2614
39600	64800	11	18	4563
		18	720	0

3.3.6 SGTR Analysis Results

The results of the analyses are presented below for the Concurrent Spike (Table 3.3-9) and for the Pre-accident Iodine Spike (Table 3.3-10).

Table 3.3-9

Dose Summary for the SGTR Concurrent Iodine Spike

Location	TEDE (rem)	Limits (rem)
EAB	9.0E-01	2.5
LPZ	9.0E-02	2.5
Millstone Unit 3 Control Room	1.3E+00	5

Table 3.3-10

Dose Summary for the SGTR Pre-accident Iodine Spike

Location	TEDE (rem)	Limits (rem)
EAB	2.1E+00	25
LPZ	1.8E-01	25
Millstone Unit 3 Control Room	3.0E+00	5

3.4 Main Steam Line Break Analysis

This section describes the methods employed and results of the Main Steam Line Break (MSLB) design basis radiological analysis. This analysis included doses associated with the releases of the radioactive material initially present in primary and secondary liquids at Technical Specification concentrations and adjusting for iodine spiking scenarios. No fuel failure is expected. Doses were calculated at the Exclusion Area Boundary (EAB), at the Low Population Zone (LPZ), and in the Millstone Unit 3 Control Room. The methodology used to evaluate the control room and offsite doses resulting from the MSLB accident is consistent with Regulatory Guide 1.183 (Reference 1) in conjunction with TEDE radiological units and limits, used ARCON96 based onsite atmospheric dispersion factors, and utilized Federal Guidance Report No. 11 and 12 (References 7 & 8, respectively) dose conversion factors.

3.4.1 MSLB Scenario Description

The Main Steam Line Break (MSLB) accident begins with a break in one of the main steam lines leading from a steam generator (affected generator) to the turbine. In order to maximize control room dose, the break is assumed to occur in the turbine building. The affected steam generator releases steam for 55.2 hours, at which time the RCS has cooled down to 212°F and release via this pathway terminates. The 55.2-hour steaming period is based on the time necessary to cooldown crediting safety grade equipment only. Also, it is expected that the affected generator will dry out in 56.3 seconds post-MSLB. Loss Of Off-site Power is assumed. As a result, the condenser is unavailable and cool down of the primary system is through the release of steam from the intact generators. The release from the intact generators continues for 18 hours through the ADVs until the RHRS can fully remove decay heat. In accordance with RG 1.183, Appendix E, two independent cases are evaluated. Case one assumes a pre-accident iodine spike, while the second case assumes a concurrent iodine spike.

3.4.2 MSLB Source Term Definition

As with the SGTR accident, the analysis of the MSLB accident indicates that no additional fuel rod failures occur as a result of the transient. Thus, radioactive material releases are determined by the radionuclide concentrations initially present in primary and secondary liquid at maximum Technical Specification limits and iodine spiking.

The Main Steam Line Break analysis uses the primary and secondary liquid source term discussed in Table 3.3-1 and the pre-accident iodine spike source term discussed in Table 3.3-2. The MSLB analysis also assumes a concurrent iodine spike as an accident initiated value 500 times the appearance rate corresponding to the Technical Specification limit for normal operation (1 uCi/gm DEQ I-131 RCS TS limit) for a period of 8 hours and that source term is listed below in Table 3.4-1.

Table 3.4-1

Concurrent Iodine Spike

Nuclide	Appearance rate for 1 uCi/gm DEQ I-131, uCi/sec	Spike = 500, MSLB Appearance Rate, uCi/sec
I131	5.24E+03	2.62E+06
I132	7.13E+03	3.57E+06
I133	1.09E+04	5.44E+06
I134	9.91E+03	4.96E+06
I135	8.77E+03	4.39E+06

3.4.3 Release Transport

The source term resulting from the radionuclides in the primary system coolant and from the iodine spiking in the primary system is transported to the steam generators by the leak-rate limiting condition for operation (1 gpm) specified in the Technical Specifications. The maximum amount of primary to secondary leakage allowed by the Technical Specifications to any one steam generator is 500 gallons per day. This leakage (500 gpd equivalent to 0.35 gpm) was assigned to the affected generator.

For the affected generator, the release pathway is assumed to pass directly into the turbine building with no credit taken for holdup, partitioning or scrubbing by the steam generator liquid. No credit is taken for any holdup or dilution in the Turbine Building. From the Turbine Building it passes to the environment and to the control room. During the first 56.3 seconds post-trip, the affected steam generator is assumed to steam dry as a result of the MSLB, releasing all of the nuclides in the secondary coolant that were initially contained in the steam generator. During the first 55.2 hours, the primary coolant is also assumed to leak into the affected steam generator at the rate of 500 GPD with all activity released unmitigated to the environment. After 55.2 hours the RCS will have cooled to below 212°F and the release via this pathway terminates. The transport model utilized for noble gases, iodine and particulates was consistent with Appendix E of Regulatory Guide 1.183.

The remainder of the 1 gpm primary side to secondary side leakage, 0.65 gpm, was assigned to 2 intact generators. This leakage continues for 18 hours until shutdown cooling is credited for decay heat removal. The third intact generator is assumed to have a failed closed atmospheric dump valve, which reduces the holdup volume to 2 generators instead of 3, but the steaming rate has not been reduced, which maximizes the release rate.

There are several nuclide transport models associated with the intact steam generators. Together, they ensure proper accounting of gross gamma, iodine and noble gas releases. The first pathway releases gross gamma activity, at the Technical Specification limit of $100/E_{bar}$, to the SG liquid volume at 0.65 gpm.

Releases of radionuclides initially in the steam generator liquid and those entering the steam generator from the primary to secondary leakage flow are released as a result of secondary liquid boiling. Due to moisture carryover, 1% of the particulates in the steam generator bulk liquid are released to the environment at the steaming rate. Radionuclides initially in the steam space do not provide any significant dose contribution and are not considered. The transport to the environment of noble gases from the primary coolant and from particulate daughters occurs without any mitigation or holdup.

The pre-accident iodine spike is modeled in the same manner as the gross gamma model previously discussed.

The concurrent iodine spike model is modeled in the same manner as the gross gamma model but the iodine spike occurs for 8 hours after which the activity remaining in the primary coolant continues to be released for the remainder of the 18 hours.

3.4.4 Determination of Atmospheric Dispersion Factors

3.4.4.1 Control Room Atmospheric Dispersion Factors

The onsite atmospheric dispersion factors were calculated by Dominion using the ARCON96 code and guidance from RG-1.194 (Reference 10) for the Control Rooms. Site meteorological data taken over the years 1997-2001 were used in the evaluations. Control room X/Q values were calculated from the closest ventilation point on the Turbine Building to the control room inlet to maximize dose. These and other control room atmospheric dispersion factors are presented in Table 1.3-4.

3.4.4.2 Offsite Atmospheric Dispersion Factors

X/Q's from the Turbine Building Ventilation Stack are used for offsite doses since the ventilation stack is located near the Turbine Building roof exhaust vents.

3.4.5 MSLB Key Analysis Assumptions and Inputs

3.4.5.1 Basic Data and Assumptions

The basic data and assumptions are listed below in Table 3.4-2. All numeric values specific to this evaluation are listed in this section. Generic data such as control room information is available in Tables 1.3-1 and 1.3-2. Steam generator mass releases and timings are a product of updated Westinghouse thermal-hydraulic analyses.

Table 3.4-2

Basic data and Assumptions for the MSLB Accident

Data / Assumption	Value
Loss of Offsite Power: • Assumed to Occur at Accident Initiation	
Release Points: Affected Steam Generator: Intact Steam Generator:	 Turbine Bldg ADVs
Iodine Partition Coefficients (PC) in Intact Steam Generators:	100
Moisture Carryover in Intact Steam Generators:	1%
Primary-to-Secondary Leakage:	Affected SG: 500 GPD Total: 1 GPM
Steam Generator Liquid Mass:	164,200 lbm
Control Room Ventilation Timing	Same as Fuel Handling Accident (Table 3.2.1)
Duration of Steam Generator Release: • Affected Steam Generator: 55.2 hours (to reach RCS temperature of 212°F) • Intact Steam Generators: 18 hours (to enter RHRS window of operation)	
Steam Release from affected Steam Generator: Initial Inventory: 0 – 56.3 seconds: 1.75E+05 lbm/min Primary-to-Secondary Leak 0 – 55.2 hours: 2.918 lbm/min (= 0.35 gpm)	

Table 3.4-2

Basic data and Assumptions for the MSLB Accident

Steam Release from Intact Steam Generators: 0 - 2 hours: 3.41E+03 lbm/ min 2 - 8 hours: 2.73E+03 lbm/ min 8 - 18 hours: 4.56E+03 lbm/ min
Millstone Unit 3 Auxiliary Feed System is available to maintain water level in intact steam generators.
Control Room Plume and Filter Shine Dose to the Millstone Unit 3 Control Room • Conservatively set the same as the LOCA analyses

3.4.5.2 MSLB Analysis Results

The total TEDE to the EAB, LPZ and Millstone Unit 3 Control Room from a Millstone Unit 3 Main Steam Line Break is summarized below for the concurrent (Table 3.4-3) and pre-accident spike (Table 3.4-4). The concurrent spike results in the highest dose consequences for both offsite and the control room. All doses are within the limits specified in Regulatory Guide 1.183 and 10 CFR 50.67.

Table 3.4-3

TEDE – Concurrent Iodine Spike

Location	TEDE (rem)	Limits (rem)
EAB	3.6E-01	2.5
LPZ	1.8E-01	2.5
Millstone Unit 3 Control Room	3.0E+00	5

Table 3.4-4

TEDE – Pre-accident Iodine Spike

Location	TEDE (rem)	Limits (rem)
EAB	9.1E-02	25
LPZ	3.6E-02	25
Millstone Unit 3 Control Room	1.2E+00	5

3.5 Locked Rotor Analysis

This section describes the methods employed and results of the Locked Rotor Accident (LRA) design basis radiological analysis. The analysis assumes failure of 7% of the fuel rods, due to Departure from Nucleate Boiling (DNB) during the accident. Doses were calculated at the Exclusion Area Boundary (EAB), at the Low Population Zone (LPZ), and in the Millstone Unit 3 Control Room. The methodology used to evaluate the control room and offsite doses resulting from the LRA included Regulatory Guide 1.183 methodology, ARCON96-based control room atmospheric dispersion factors, and Federal Guidance Reports (FGR) No. 11 and 12 dose conversion factors.

3.5.1 Locked Rotor Scenario Description

The Locked Rotor Accident (LRA) begins with instantaneous seizure of the reactor coolant pump rotor under 4 loop operation. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer, which results in assumed fuel damage due to Departure from Nucleate Boiling (DNB). Although there is no increase in the leak rate of primary coolant to the secondary side during the LRA, a larger amount of activity (from the failed fuel) is transported to the secondary side via any pre-existing leaks in the steam generators.

A turbine trip and coincident loss of offsite power are incorporated into the analysis. This results in a release through a stuck open steam generator atmospheric dump valve (ADV) and a parallel release from the intact steam generators. The stuck-open ADV represents the assumed, single, active failure. Consistent with current licensing bases, operator action is credited with closure of the ADV after 20 minutes.

3.5.2 Locked Rotor Source Term Definition

The core source term used in the Locked Rotor Analysis is taken from Table 3.1-3. Analyses are based on 7% of the gap activity being released.

3.5.3 Release Transport

The release scenario uses the Technical Specification primary to secondary leakage limits of 1 gpm total and 500 gpd from the affected steam generator. The release from the affected steam generator continues for 20 minutes until operator action isolates that release pathway.

The balance of the 1 gpm limit (0.65 gpm) is released from the intact steam generators over the course of 18 hours until shutdown cooling can be implemented to fully remove decay heat. At this point the release from the intact steam generators is terminated when the operator closes the ADVs.

The RADTRAD-NAI computer code (Reference 3) is used to model the time dependent transport of radionuclides, from the primary to secondary side and out to the environment via ADVs.

3.5.4 Determination of Atmospheric Dispersion Factors (X/Q)

3.5.4.1 Control Room Atmospheric Dispersion Factors (X/Q)

The Millstone Unit 3 Control Room X/Q values were calculated by using the ARCON96 code and guidance from RG 1.194. The control room X/Q values are calculated at the Main Steam Valve Building. No credit is taken for the thermal plume rise. These X/Q values are given in Table 1.3-4.

3.5.4.2 Offsite X/Q

The EAB and LPZ X/Q values used in the Locked Rotor analysis are the same as those used in the LOCA analysis listed in Table 1.3-3.

3.5.5 Locked Rotor Analysis Assumptions and Key Parameters

3.5.5.1 Basic Data and Assumptions

All numeric values specific to this evaluation are listed in Table 3.5-1. Generic data such as control room information is available in Tables 1.3-1 and 1.3-2.

Table 3.5-1

Basic Data and Assumptions for the LRA

Parameter / Assumption	Value
Fuel Damage	7% Fuel Failure
Radial Peaking Factor:	1.7
Primary to Secondary Leak Rate: (Technical Specifications)	0.35 gpm (affected SG) 0.65 gpm (intact SGs)
Release from secondary side is coincident with loss of off-site power	
Release Points:	Steam Generator Atmospheric Dump Valve (ADV)
Credited Operator Actions a. Operator Action to close ADV after 20 minutes b. Operator actions commensurate with cooldown using only safety grade equipment	These operator actions are currently part of the Licensing Bases.
Iodine Chemical Form Released from Steam Generators to Environment:	Elemental 97% Organic 3%
Fraction of Fission Product Inventory in Gap	Halogens: 0.8 Noble Gases: 0.10 Alkali Metals: 0.12
Iodine Partitioning in Intact Steam Generator	100
Intact Steam Generator Tube Uncovery	No tube bundle uncovery assumed.

Table 3.5-1

Basic Data and Assumptions for the LRA

Parameter / Assumption	Value
Affected Steam Generator Tube Uncovery • Affected steam generator goes dry, immediately • 100% flashing is assumed • Conservative assumption because no feedwater is credited to this generator and the mass of water pre-existing in the generator, 9.722E+04 lb • Contents discharged out the ADV at 820,000 lb/hr • SG dries out in approximately 7 minutes.	
Release Duration: • Intact Steam Generators – 18 hours based on cooldown using only safety grade equipment • Affected Steam Generator – 20 minutes based on operator action	
Total Steam Flows to Atmosphere from 3 Intact Steam Generators	0 - 2 hours: 251,000 lbm 2 - 8 hours: 1,031,000 lbm 11-18 hours: 1,915,359 lbm
Mass Flow Rates from 3 Intact Steam Generators	0 - 2 hours: 1.255E+05 lbm/hr 2 - 8 hours: 1.718E+05 lbm/hr 8 - 11 hours: 2.736E+05 lbm/hr 11 - 18 hours: 2.736E+05 lbm/hr
Moisture carryover in intact Steam Generators:	1%
Initial Steam Generator Liquid Mass	4.414E+07 grams / Steam Generator
Millstone Unit 3 Steam Generator ADV Maximum Flow Rate	820,000 lbm/hr @1140 psia
Control Room Ventilation Timing	Same as the Fuel Handling Accident (Table 3.2-1)
Assumption: • Dose consequences are from release of initial secondary side liquid • Dose consequences from the release of steam is not significant	
Control Room Plume and Filter Shine Dose to the Millstone Unit 3 Control Room • Conservatively set the same as the LOCA analyses	

3.5.6 Locked Rotor Analysis Results

The results of the design basis Locked Rotor analysis are presented in Table 3.5-2. These results report the calculated dose for the worst 2-hour interval (EAB), and for the assumed 30-day duration of the event for the control room and the LPZ. As stated in Table 3.5-1, plume and filter shine to the control room are conservatively based on LOCA results and will be used here. The doses are calculated with the TEDE methodology, and are compared with the applicable acceptance criteria specified in 10 CFR 50.67 and Regulatory Guide 1.183.

Table 3.5-2

TEDE Results for the Locked Rotor Accident

Location	TEDE (rem)	Limits (rem)
EAB	2.3E+00	2.5
LPZ	3.7E-01	2.5
Millstone Unit 3 Control Room	3.2E+00	5

3.6 RCCA Ejection Accident Analysis

This section describes the evaluation of TEDE at the EAB, LPZ and MP3 Control Room from a Millstone Unit 3 Rod Control Cluster Assembly (RCCA) Ejection Accident (REA) using the AST. Two release cases are considered. The first case is a release into the containment. The second release is a release into the primary coolant, which is released through the secondary system.

3.6.1 RCCA Ejection Accident Scenario Description

This accident is defined as the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of a RCCA and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

3.6.2 RCCA Ejection Accident Source Term Definition

The core source term used in the RCCA Ejection Accident Analysis are taken from Table 3.1-3. The release of the core source term is adjusted for the fraction of fuel rods assumed to fail during the accident and the fractions of core inventory assumed to be in the pellet-to-clad gap.

The analysis is based on the assumption that 10% failed fuel and 0.25% melted fuel occurs during a RCCA Ejection Accident.

3.6.3 Release Transport

Two release paths are considered for the REA: containment leakage and the secondary system.

The containment release transport assumptions and methodology are similar to the LOCA and can be found in section 3.1.5, with a few exceptions. The exceptions are:

1) The core release fractions are based on Appendix H of R.G. 1.183. The core release fractions are 0.010625 for halogens and 0.0125 for noble gases based on the consequences of 10% failed fuel and 0.25% melted fuel.
2) Containment sprays do not initiate due to a REA. Therefore there are no consequences from ECCS leakage and RWST backleakage.
3) Natural deposition in the containment is not assumed.
4) Containment leak rate is reduced by 50% at 24 hours for both offsite and control room analyses.
5) The safety injection signal is initiated 2 minutes after a REA. Therefore the control room is not isolated until 2 minutes 10 seconds following a REA.

The second release path is via the secondary system. The activity in the secondary system release is based on Appendix H of RG 1.183. The core release fractions are 0.01125 for halogens and 0.0125 for noble gases based on the consequences of 10% failed fuel and 0.25% melted fuel. The iodines released from the steam generators are assumed to be 97% elemental and 3% organic. The primary-to-secondary leak rate of 1 gpm, which is specified in the technical specifications, exists until shutdown cooling is in operation and release from the steam generators terminate. All noble gas radionuclides released to the secondary system are released to the environment without reduction or mitigation. The condenser is not available due to a loss of offsite power. A partition coefficient for iodine of 100 is assumed in the steam generators.

The primary-to-secondary leak occurs during the first 1,200 seconds of the REA (until primary system pressure is less than secondary side system pressure). Steam generator mass releases are a product of updated Westinghouse thermal-hydraulic analyses. The steam released via the safeties/ADVs during the REA and subsequent cooldown is listed in Table 3.6-1.

3.6.4 Determination of Atmospheric Dispersion Factors (X/Q)

3.6.4.1 Control Room Atmospheric Dispersion Factors (X/Q)

The Millstone Unit 3 Control Room X/Q values are given in Table 1.3-4.

3.6.4.2 Offsite Atmospheric Dispersion Factors (X/Q)

The EAB and LPZ X/Q values used in the REA analysis are the same as those used in the LOCA analysis and are listed in Table 1.3-3

3.6.5 RCCA Ejection Accident Analysis Assumptions and Key Parameters

3.6.5.1 Basic Data and Assumptions

The basic data and assumptions are listed below in Table 3.6.1. Generic data, such as control room information, is available in Tables 1.3-1 and 1.3-2.

Table 3.6-1

Basic Data and Assumptions for the REA

Parameter / Assumption	Value
Core Release Fractions for Breached Fuel: (Appendix H of Reference 1)	10% of noble gasses and iodines in the gap
Core Release Fractions for Melted Fuel: (Appendix H of Reference 1)	100% noble gasses and 25% iodines
Percentage of Failed Fuel following a REA:	10%
Percentage of Melted Fuel following a REA:	0.25%
Core Release Fractions for Secondary Side Release: (Appendix H of Reference 1):	100% noble gasses and 50% iodines in the fraction released to the reactor coolant
Safety Injection (SI) Signal Initiated after a REA:	2 minutes

Table 3.6-1

Basic Data and Assumptions for the REA

Parameter / Assumption	Value
Iodine Chemical Form Released from the Steam Generators to the Environment: (Reference 1)	3% Organic Iodide 97% Elemental Iodine
Total reactor-to-secondary leakage through all steam generators: (Technical Specifications, Section 3.4.6.2.c)	1 gpm
Time for primary system pressure to fall below secondary system pressure:	1,200 seconds
Duration of steam releases:	18 hours
Steam released from t=0 to 1200 seconds (primary system depressurization):	200,000 lbm
Steam released from 2 – 11 hours:	1.547E+06 lbm
Steam released from 11 – 18 hours:	1.916E+06 lbm

3.6.6 RCCA Ejection Accident Analysis Results

The total TEDE to the EAB, LPZ, and the Millstone Unit 3 Control Room from a Millstone Unit 3 RCCA Ejection Accident (REA) is summarized below for the containment pathway (Table 3.6-2) and the secondary side release pathway (Table 3.6-3). The dose to the EAB and LPZ is less than the 6.3 rem TEDE limit stated in 10CFR50.67 and Regulatory Guide 1.183. The EAB dose represents the worst 2-hour dose for each release pathway. The dose to the Millstone Unit 3 Control Room is less than the 5 rem TEDE limit specified in 10CFR50.67 and Regulatory Guide 1.183.

Table 3.6-2

TEDE from a Millstone Unit 3 REA (containment)

Location	TEDE (rem)
EAB	8.7E-01
LPZ	4.8E-01
Millstone Unit 3 Control Room	8.3E-01

Table 3.6-3

TEDE from a Millstone Unit 3 REA (secondary side)

Location	TEDE (rem)
EAB	1.2E-01
LPZ	1.5E-02
Millstone Unit 3 Control Room	5.3E-02

4.0 ADDITIONAL DESIGN BASIS CONSIDERATIONS

In addition to the explicit evaluation of radiological consequences that had direct impact from the changes associated with implementing the AST, other areas of plant design were also considered for potential impacts. The evaluation of these additional design areas is documented below.

4.1 Impact Upon Equipment Environmental Qualification

In the Federal Register notice issuing the final rule for use of alternative source terms at operating reactors (Reference 38), the NRC stated that it will evaluate this issue as a generic safety issue to determine whether further regulatory actions are justified. This issue was subsequently designated as Issue 187: The Potential Impact of Postulated Cesium Concentration on Equipment Qualification. Further guidance is provided in SECY-99-240 (Reference 39), which transmitted the final AST rule changes for the Commission's approval. The following is stated in the 'Discussion' section, regarding evaluation of the equipment qualification issue before its final resolution:

"In the interim period before final resolution of this issue, the staff will consider the TID-14844 source term to be acceptable in reanalyses of the impact of proposed plant modifications on previously analyzed integrated component doses regardless of the accident source term used to evaluate offsite and control room doses."

In NUREG-0933, Supplement 25 (Reference 40), the NRC staff reported its conclusions concerning the assessment of Issue 187. The staff concluded that there was no clear basis to require that the equipment qualification design basis be modified to adopt the AST. It was stated that there would be no discernable risk reduction from such a requirement. This issue was thus dropped from further pursuit. Consistent with this guidance, no further evaluation of this issue is presented in support of implementing the AST for Millstone Unit 3. The existing equipment qualification analyses, which are based upon the TID-14844 source term, are considered acceptable.

4.2 Risk Impact of Proposed Changes Associated with AST Implementation

Implementation of the AST is of benefit to licensees because of the potential to obtain relaxation in specific safeguard systems operability or surveillance requirements, since such changes can reduce regulatory burden and streamline operations. Such changes are warranted if they can be pursued without creating an unacceptable impact upon plant risk characteristics as compared with the existing system licensing and operational basis. The proposed changes associated with implementation of the AST for Millstone Unit 3 have been considered for their risk effects. A discussion of these considerations is presented below.

The proposed changes are presented here for convenience; these changes are described in report sections 2.2 through 2.6:

a. The definition of Dose Equivalent 1-131 in Section 1.10 of the Technical Specifications Definitions is revised to reference Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, " 1989, as the source of thyroid dose conversion factors (Reference 7).

b. Change Technical Specification 3/4.7.7, "Control Room Emergency Air Filtration System," Surveillance Requirements c.2 and d to reflect a methyl penetration less than or equal to 5% for the Control Room Emergency Air Filtration System filters instead of 2.5%.

c. Delete Technical Specification 3/4.7.8, "Control Room Envelope Pressurization System." The Control Room Envelope Pressurization System is no longer credited in the accident analyses described in evaluation.

d. Change the leakage rate acceptance criteria for all penetrations that are Secondary Containment bypass leakage paths in Section 6.8.4.f, "Containment Leakage Testing Program," from $\leq 0.042\ L_a$ to $\leq 0.06\ L_a$.

Item a – This change allows the use of dose conversion factors from FGR 11. These dose conversion factors have been previously found to be acceptable for use in dose calculations. This change has no impact upon plant risk from severe accident scenarios.

Item b - Item d – The changes to the specified test acceptance criteria for the Control Room Emergency Air Filtration System ensure that system performance remains consistent with the assumptions of the accident analyses. Although the changes represent relaxation of acceptance criteria, this is judged to represent negligible risk impact since the most risk-significant scenarios are for accident sequences in which filtration systems have already lost power or are ineffective in reducing radioactive releases.

In addition, the risk associated with modification and/or elimination of such filtration systems was evaluated during the NRC's rebaselining study (Reference 37). Reference 37 reported that the effect on overall risk from filtration system modifications was small. This effect was attributed to the fact that filtration systems, which require electrical power for operation, will already not be functional for certain risk-significant accident sequences (e.g., station blackout).

It is concluded that the proposed changes in test acceptance criteria will produce negligible impact upon overall plant risk for such accident sequences.

Items c and d – These changes remove the pressurization of the control room early in the accident sequence and increase the leakage rate acceptance criteria for all penetrations that are Secondary Containment bypass leakage paths. These changes are accomplished while maintaining calculated doses to the Control Room operators, the EAB and the LPZ within the TEDE limits of 10CFR50.67. These limits have been judged to be acceptable consequences. These changes are expected to have negligible impact on plant risk associated with severe accident sequences.

It is concluded that the proposed changes associated with AST implementation for Millstone Unit 3 will have insignificant effect upon the risk associated with severe accidents. This is primarily due to the fact that the risk significant accident sequences involve the failure of systems or structures (e.g., containment) that are not impacted by the relatively minor operational changes proposed herein.

4.3 Impact Upon Emergency Planning Radiological Assessment Methodology

This application of the AST for Millstone Unit 3 replaces the existing design basis source term with the source term defined in RG 1.183. The MIDAS model that is employed for emergency planning radiological assessments includes definitions of source terms for various design basis accidents. Calculated results from MIDAS are used in various emergency preparedness processes. The basis of the existing source term definitions in the MIDAS calculations will be evaluated to determine: 1) the manner in which the source terms used in emergency preparedness activities rely upon the design basis event source term definition and 2) what specific changes may be warranted in the emergency preparedness source terms and their detailed usage. This assessment of potential impact will also include radiation monitor setpoint calculations for accident high range monitors, which use data input similar to MIDAS.

5.0 Conclusions

The alternative source term defined in Regulatory Guide 1.183 has been incorporated into the reanalysis of radiological effects from six key accidents for Millstone Unit 3. This amendment request represents a full implementation of the alternative source term, making RG 1.183 the licensing basis source term for assessment of design basis events. The analysis results from the reanalyzed events meet all of the acceptance criteria as specified in 10 CFR 50.67 and RG 1.183.

6.0 References

1. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," USNRC, Office of Nuclear Regulatory Research," July 2000.

2. 10 CFR 50.67, "Use of Alternative Source Terms at Operating Reactors," Final Rule, in Federal Register No. 64, p. 71990, December 23, 1999.

3. RADTRAD-NAI 1.1(QA), Numerical Applications Inc.

4. NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," USNRC, June 1997, S.L. Humphreys et al.

5. NUREG/CR-6331, Rev. 1, "Atmospheric Relative Concentrations in Building Wakes, ARCON96," USNRC, 1997.

6. Computer Code SCALE 4.4a Version 1 Mod 0.

7. Technical Information Document TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," United States Atomic Energy Commission, 1962.

8. Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA 520/1-88-020, Environment Protection Agency, 1988.

9. Federal Guidance Report No. 12, "External Exposures to Radionuclides in Air, Water and Soil," EPA 420-r-93-081, Environmental Protection Agency, 1993.

10. Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, June 2003

11. NUREG-0800, "Standard Review Plan," Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," U.S. Nuclear Regulatory Commission, Rev. 2, December 1988.

12. NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," July 01, 1996.

13. NRC Letter dated September 16, 2002, Victor Nerses (USNRC) to J. A. Price (DNC), "Millstone Power Station, Unit 3 – Issuance of Amaendment Re: Revised Final Safety Analysis Report Licensing Masis for the Post-Accident Operation of the Supplementary Leakage Collection and Release System (TAC No. MB3700)."

14. NRC Letter dated November 25, 2002, Victor Nerses (USNRC) to J. A. Price (DNC), "Millstone Power Station, Unit No. 3 – Correction to Safety Evaluation (SE) for Amendment No. 211 (TAC No. MB3700)."

15. BNP-100, Iodine Removal from Containment Atmospheres by Boric Acid Spray, July 1970.

16. "Radiological Health Handbook", January 1970.

17. WCAP-7828, "Radiological Consequences of a Fuel Handling Accident," December 1997.

18. SCALE Manual, NUREG/ CR-0200, Rev. 6, Volume 3, Section M8, "Standard Composition Library

19. Northeast Nuclear Energy Company Letter dated June 6, 1998, M. H. Brothers (NNECo) to USNRC, "Millstone Nuclear Power Station, Unit No. 3, Proposed License Amendment Request, SLCRS Bypass Leakage, (PLAR 3-98-5)."

20. Northeast Nuclear Energy Company Letter dated April 5, 1999, Raymond P. Necci (NNECo) to USNRC, "Millstone Nuclear Power Station, Unit No. 3, License Amendment Related to the Supplementary Leakage Collection and Release System (PLAR 3-98-5), Response to Request for Additional Information."

21. Northeast Nuclear Energy Company Letter dated April 7, 2000, Raymond P. Necci (NNECo) to USNRC, "Millstone Nuclear Power Station, Unit No. 3, License Amendment Related to the Supplementary Leakage Collection and Release System (PLAR 3-98-5), Supplemental Information."

22. Northeast Nuclear Energy Company Letter dated April 9, 2000, M.H. Brothers (NNECo) to USNRC, "Millstone Nuclear Power Station, Unit No. 3, License Amendment Related to the Supplementary Leakage Collection and Release System (PLAR 3-98-5), Supplemental Information."

23. Northeast Nuclear Energy Company Letter dated July 31, 2000, Raymond P. Necci (NNECo) to USNRC, "Millstone Nuclear Power Station, Unit No.

3, License Amendment Related to the Supplementary Leakage Collection and Release System (PLAR 3-98-5), Supplemental Information."

24. Northeast Nuclear Energy Company Letter dated September 28, 2000, Raymond P. Necci (NNECo) to USNRC, "Millstone Nuclear Power Station, Unit No. 3, License Amendment Related to the Supplementary Leakage Collection and Release System (PLAR 3-98-5), Supplemental Information."

25. Northeast Nuclear Energy Company Letter dated March 19, 2001, Raymond P. Necci (NNECo) to USNRC, "Millstone Nuclear Power Station, Unit No. 3, License Amendment Related to the Supplementary Leakage Collection and Release System (PLAR 3-98-5), Supplemental Information."

26. Dominion Nuclear Connecticut, Inc. Letter dated June 11, 2001, Raymond P. Necci (DNC) to USNRC, "Millstone Nuclear Power Station, Unit No. 3, License Amendment Related to the Supplementary Leakage Collection and Release System (PLAR 3-98-5), Clarification of Proprietary Information."

27. Dominion Nuclear Connecticut, Inc. Letter dated September 21, 2001, J. Alan Price (DNC) to USNRC, "Millstone Power Station, Unit No. 3, License Amendment Related to the Supplementary Leakage Collection and Release System (PLAR 3-98-5)."

28. Dominion Nuclear Connecticut, Inc. Letter dated December 20, 2001, Raymond P. Necci (DNC) to USNRC, "Millstone Nuclear Power Station, Unit No. 3, License Amendment Related to the Supplementary Leakage Collection and Release System (PLAR 3-98-5), Supplemental Information."

29. USNRC Letter dated November 4, 1999, John A. Nakoski (USNRC) to R.P. Necci (NNECo), Millstone Nuclear Power Station, Unit No. 3 – Issuance of Amendment Re: Reactor Water Storage Tank Back Leakage (TAC No. MA1749)."

30. Northeast Nuclear Energy Company Letter dated May 7, 1998, M. H. Brothers (NNECo) to USNRC, "Millstone Nuclear Power Station, Unit No. 3, Proposed License Amendment Request, Refueling Water Storage Tank Back Leakage (PLAR 3-98-3)."

31. Northeast Nuclear Energy Company Letter dated January 22, 1999, Martin L. Bowling (NNECo) to USNRC, "Millstone Nuclear Power Station, Unit

No. 3, License Amendment Request, Refueling Water Storage Tank Back Leakage (PLAR 3-98-3), Request for Additional Information."

32. NSAL-00-004, Westinghouse Nuclear Safety Advisory Letter dated March 7, 2000, , Non-conservatisms in Iodine Spiking Calculations.

33. Westinghouse Report WCAP-13132, "The Effect of Steam Generator Tube Uncovery on Radioiodine Release," January 1992. (Proprietary)

34. Dominion Nuclear Connecticut Letter dated March 4, 2003, J. Alan Price (DNC) to USNRC, "Millstone Power Station, Unit No. 3, Licensing Basis Document Change Request (LBDCR) 3-01-03, Selective Implementation of the Alternative Source Term – Fuel Handling Accident Analysis."

35. V. Nerses, USNRC to D. Christian, "Millstone Station Unit No. 3 – Issuance of Amendment Re: Selective Implementation of Alternative Source Term (TAC No. MB8137), dated March 17, 2004.

36. Regulatory Guide 1.145, Revision 01, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," February 1983.

37. SECY-98-154, "Results of the Revised (NUREG-1465) Source Term Rebaselining for Operating Reactors," June 30, 1998.

38. "Use of Alternative Source Terms at Operating Reactors," Final Rule, in Federal Register No. 64, p. 71990, December 23, 1999.

39. SECY-99-240, "Final Amendment to 10 CFR Parts 21, 50, and 54 and Availability for Public Comment of Draft Regulatory Guide DG-1081 and Draft Standard Review Plan Section 15.0.1 Regarding Use of Alternative Source Terms at Operating Reactors," October 5, 1999.

40. NUREG-0933, "A Prioritization of Generic Safety Issues," Supplement 25, June 2001.

7.0 Technical Specification And Bases Change

The following Technical Specifications for Millstone Unit 3 are revised as noted below to reflect implementation of the NUREG-1465 alternative source term (AST) as the Design Basis Source Term. The AST implementation analyses provide justification for the following changes to the Millstone Unit 3 Technical Specifications and Technical Specification Bases:

a. The definition of Dose Equivalent 1-131 in Section 1.10 of the Technical Specifications Definitions is revised to reference Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989, as the source of thyroid dose conversion factors (Reference 8). The AST implementation analyses, as described in Section 3, use the thyroid dose conversion factors listed in Federal Guidance Report No. 11 (FGR 11) instead of those listed in NRC Regulatory Guide 1.109, Revision 1.

b. Change Technical Specification 3/4.7.7, "Control Room Emergency Air Filtration System," Surveillance Requirements c.2 and d to reflect a methyl iodide penetration less than or equal to 5% for the Control Room Emergency Air Filtration System filters instead of 2.5%. The AST implementation analyses assumed Charcoal Filter Efficiencies (%) for Control Room Filtered Recirculation and Intake as 90 aerosol, 90 elemental, and 70 organic. Millstone Unit 3 adopts ASTM D3803-89 as the standard test method for Nuclear-Grade activated carbon and to determine the acceptance criteria of methyl iodide penetration. The allowable penetration in accordance with ASTM D3803-89 corresponding to 90% methyl iodide efficiency for charcoal credited in Millstone Unit 3 AST implementation analyses is 5% using a safety factor of 2.

c. Delete Technical Specification 3/4.7.8, "Control Room Envelope Pressurization System." The Control Room Envelope Pressurization System is no longer credited in the accident analyses described in the AST implementation analyses. In accordance with AST implementation analyses, the requirements contained in this Specification do not meet any of 10 CFR 50.36(c)(2)(ii) criteria on items for which Technical Specifications must be established. This can be justified as follows:

Justification:

The Control Room Envelope Pressurization System provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. The Control Room Envelope Pressurization System consists of two banks of air bottles with its associated piping, instrumentation, and controls. Each bank is capable of providing the control room area with one-hour of air following any event with the potential for radioactive releases. During normal operations, the Control Room Envelope Pressurization System is required to be on standby. The Control Room Envelope Pressurization System is required to operate during post accident operations to ensure the control room will remain habitable during and following accident conditions.

Technical Specification 3/4.7.8 provide operability requirement, associated actions and surveillance requirement for the Control Room Envelope Pressurization System. The Control Room Envelope Pressurization System is no longer credited in the accident analyses described in the AST implementation analyses as described in Section 3. Additionally, this specification does not meet any of the criteria of 10 CFR 50.36(c)(2)(ii).

10 CFR 50.36(c)(2)(ii) contains the requirements for items that must be in Technical Specifications. This regulation provides four (4) criteria that can be used to determine the requirements that must be included in the Technical Specifications.

Criterion 1

Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

This Specification provides for the criteria used in determining operability of the Control Room Envelope Pressurization System. This specification does not cover installed instrumentation that is used to detect, and indicate in the control room, a significant degradation of the reactor coolant pressure boundary. Therefore this specification does not satisfy criterion 1.

Criterion 2

A process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

This Specification provides for the criteria used in determining operability of Control Room Envelope Pressurization System. This specification does not cover a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore this specification does not satisfy Criterion 2.

Criterion 3

A System, Structure, or Component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

This Specification requires Control Room Envelope Pressurization System to be OPERABLE in MODES 1 through 6, and during fuel movement within containment or spent fuel pool. The AST implementation analyses, as described in Section 3, do not assume the Control Room Envelope Pressurization System available in these analyses. This assumption will provide the basis for removing Technical Specification 3/4.7.8 because it will no longer be credited in the accident analysis. Therefore, this feature does not cover a System, Structure, or component that is part of the primary success path 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. This Specification does not satisfy Criterion 3.

Criterion 4

A SSC which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The Specification, which provides the criteria used in determining operability of the Control Room Envelope Pressurization System, has not been shown to be risk significant to public health and safety by either operating experience or probabilistic safety assessment. The subject system is not credited to ensure radiological dose criteria for the EAB, LPZ, or control room is met. With the changes proposed in this license amendment request, this requirement no longer covers a SSC which requires risk review/unavailability monitoring. This Specification does not satisfy Criterion 4.

In conclusion, the proposed changes to this specification (3/4.7.8) do not cover plant equipment which is credited to function in the event of a DBA. Additionally, the requirements contained in this Specification do not meet any of 10 CFR 50.36(c)(2)(ii) criteria regarding items for which Technical Specifications must be established. Therefore, the proposed change to delete Technical Specification 3/4.7.8 is consistent with regulation and is safe.

d. Change the leakage rate acceptance criteria for all penetrations that are secondary containment bypass leakage paths in Section 6.8.4.f, "Containment Leakage Testing Program," from $\leq 0.042\ L_a$ to $\leq 0.06\ L_a$. The AST implementation analyses, as described in Section 3, assume leakage rate acceptance criteria for all penetrations that are secondary containment bypass leakage paths to be $\leq 0.06\ L_a$.

e. Index pages x and xiv are revised to reflect the deletion of Technical Specification 3/4.7.8 and the corresponding Bases.

The associated Bases changes are provided for information only. The Technical Specification Bases will be revised in accordance with the Technical Specification Bases Control Program (Section 6.18), following approval of the AST license amendment.

7.1 Specific Technical Specification Changes

In this section deleted text is omitted and inserted text is underlined in the To portion of each revision. The Bases changes are included with each technical specification that is changed.

7.1.1 Definitions

<u>**Revise the current Definition of DOSE EQUIVALENT I-131 from:**</u>

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, 1-133, I-134, and 1-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in NRC Regulatory Guide 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I."

To:

DOSE EQUIVALENT 1-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-13 1, 1-132, 1-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Federal Guidance No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion."

7.1.2 Technical Specification 3/4.7.7, "Control Room Emergency Air Filtration System"

Revise Surveillance Requirement c.2 from:

Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,* shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F), a relative humidity of 70%, and a face velocity of 54 ft/min; and

To:

Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,* shows the methyl iodide penetration less than or equal to 5.0% when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F), a relative humidity of 70%, and a face velocity of 54 ft/min; and

Revise Surveillance Requirement d from:

d. After every 720 hours of charcoal absorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,* shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F), and a relative humidity of 70%, and a face velocity of 54 ft/min.

To:

d. After every 720 hours of charcoal absorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,* shows the methyl iodide penetration less than or equal to <u>5.0%</u> when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F), and a relative humidity of 70%, and a face velocity of 54 ft/min.

7.1.3 Bases 3/4.7.7, "Control Room Emergency Air Filtration System"

Revise Background, Post Accident Operation, Item 2 from

Delete items 2. and 3., and renumber item 4. as 2.

Revise Background, Applicable Safety Analysis

Delete "except a Fuel Handling Accident, the radiation exposure to personnel occupying the control room shall be 5 rem or less whole body, or its equivalent for the duration of the accident, consistent with the requirements of General Design Criterion 19 of Appendix "A,"10 CFR 50. For a Fuel Handling Accident."

Revise 4.7.7.e.2, first paragraph on Page B 3/4 7-16 from

During the first hour, the control room pressurization system creates and maintains the positive pressure in the control room. This capability is verified by Surveillance Requirement 4.7.8.C, independent of Surveillance Requirement 4.7.7.e.2. A CBI signal will automatically align an operating filtration system into the recirculation mode of operation due to the isolation of the air supply line to the filter.

To:

A CBI signal will automatically align an operating filtration system into the recirculation mode of operation due to the isolation of the air supply line to the filter.

7.1.4 Technical Specification 3/4.7.8, "Control Room Envelope Pressurization System"

Delete Technical Specification 3/4.7.8. The text in Technical Specification pages 3/4 7-18 and 3/4 7-19 is replaced with

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7.1.5 Bases 3/4.7.8, "Control Room Envelope Pressurization System"

Delete Bases 3/4.7.8. The text in Bases pages B 3/4 7-17, 7-18, 7-19, 7-20, 7-20a, 7-21 and 7-22 will be replaced with

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7.1.6 Section 6.8.4.f, "Containment Leakage Testing Program"

Revise Section f.1 from

Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.042\ L_a$ for all penetrations that are Secondary Containment bypass leakage paths, and $< 0.75\ L_a$ for Type A tests;

To:

Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the combined Type B and Type C tests, and $\underline{\leq 0.06}\ L_a$ for all penetrations that are Secondary Containment bypass leakage paths, and $< 0.75\ L_a$ for Type A tests;

7.1.7 Additional Bases Changes

1. Bases 3/4.6.1.1, "Containment Integrity"
   Change "10 CFR Part 100" to "10 CFR 50.67", and "GDC 19" to "Regulatory Guide 1.183."

2. Bases 3/4.6.6.2, "Secondary Containment"
   Change "10 CFR Part 100" to "10 CFR 50.67."

3. Bases 3/4.7.1.4, "Specific Activity"
   Replace "a small fraction of 10 CFR Part 100" with "10 CFR 50.67 and Regulatory Guide 1.183."

4. Bases 3/4.9.10 and 3/4.9.11 "Water Level – Reactor Vessel and Storage Pool"
   Add "at least" and delete "10%."

ENCLOSURE 1 to ATTACHMENT 1

ELECTRONIC COPY

OF

SITE METEOROLOGICAL DATA TAKEN OVER THE

YEARS 1997-2001

and

CALCULATION OF ONSITE

X/Q VALUES TO THE MILLSTONE UNIT 3 CONTROL

ROOM USING ARCON96

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATION CHANGES
IMPLEMENTATION OF ALTERNATE SOURCE TERM

MARK-UP PAGES OF TECHNICAL SPECIFICATIONS CHANGES

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE containment automatic isolation valve system*, or

2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are opened under administrative control as permitted by Specification 3.6.3.

b. All equipment hatches are closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of the Containment Leakage Rate Testing Program, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATIONS

1.9 CORE ALTERATIONS shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ~~NRC Regulatory Guide 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I."~~

Replace with Insert A

* In MODE 4, the requirement for an OPERABLE containment isolation valve system is satisfied by use of the containment isolation actuation pushbuttons.

Insert A to Page 1-2

Federal Guidance No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion."

For Information only

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM   February 20, 2002

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent Control Room Emergency Air Filtration Systems shall be OPERABLE.* #

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6.
During fuel movement within containment or the spent fuel pool.

ACTION:

MODES 1, 2, 3 and 4:

a. With one Control Room Emergency Air Filtration System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both Control Room Emergency Air Filtration Systems inoperable, except as specified in ACTION c., immediately suspend the movement of fuel assemblies within the spent fuel pool. Restore at least one inoperable system to OPERABLE status within 1 hour or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

c. With both Control Room Emergency Air Filtration Systems inoperable due to an inoperable Control Room boundary, immediately suspend the movement of fuel assemblies within the spent fuel pool and restore the Control Room boundary to OPERABLE status within 24 hours or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6, and fuel movement within containment or the spent fuel pool:

d. With one Control Room Emergency Air Filtration System inoperable, restore the inoperable system to OPERABLE status within 7 days. After 7 days, either initiate and maintain operation of the remaining OPERABLE Control Room Emergency Air Filtration System in the recirculation mode of operation, or immediately suspend CORE ALTERATIONS and the movement of fuel assemblies.

e. With both Control Room Emergency Air Filtration Systems inoperable, or with the OPERABLE Control Room Emergency Air Filtration System required to be in the recirculation mode by ACTION d. not capable of being powered by an OPERABLE emergency power source, immediately suspend CORE ALTERATIONS and the movement of fuel assemblies.

---

* The requirements of Surveillance Requirement 4.7.7.e.2 do not apply during pressure testing of the Cable Spreading Room. This exception is valid until the first entry into MODE 4 following the completion of refueling operations associated with the seventh Refueling Outage.

The Control Room boundary may be opened intermittently under administrative control.

4.7.7 Each Control Room Emergency Air Filtration System shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 95°F;

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying a system flow rate of 1,120 cfm ±20% and that the system operates for at least 10 continuous hours with the heaters operating;

c. At least once per 24 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1) Verifying that the system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Position C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revisions 2, March 1978,* and the system flow rate is 1,120 cfm ±20%;

   2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,* shows ~~the methyl~~ 5.0 iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F), a relative humidity of 70%, and a face velocity of 54 ft/min; and

   3) Verifying a system flow rate of 1,120 cfm ±20% during system operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,* shows ~~the methyl iodide~~ 5.0 penetration less than or equal to 2.5% when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F), and a relative humidity of 70%, and a face velocity of 54 ft/min.

e. At least once per 24 months by:

   1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.75 inches Water Gauge while operating the system at a flow rate of 1,120 cfm ±20%;

2) 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

3) Verifying that the heaters dissipate 9.4 ±1 kW when tested in accordance with ANSI N510-1980.

f. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 1120 cfm ±20%; and

g. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 1120 cfm ±20%.

*ANSI N510-1980 shall be used in place of ANSI N510-1975 referenced in Regulatory Guide 1.52, Revision 2, March 1978.

3/4.7.8 CONTROL ROOM ENVELOPE PRESSURIZATION SYSTEM

February 20, 2002

LIMITING CONDITION FOR OPERATION

3.7.8 Two independent Control Room Envelope Pressurization Systems shall be OPERABLE.* #

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6.
During fuel movement within containment or the spent fuel pool.

ACTION:

MODES 1, 2, 3, and 4:

a. With one Control Room Envelope Pressurization System inoperable restore the system to OPERABLE status within 7 days or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

b. With both Control Room Envelope Pressurization Systems inoperable, except as specified in ACTION c. or ACTION d., immediately suspend the movement of fuel assemblies within the spent fuel pool. Restore at least one inoperable system to OPERABLE status within 1 hour or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

c. With both Control Room Envelope Pressurization Systems inoperable due to an inoperable Control Room boundary, immediately suspend the movement of fuel assemblies within the spent fuel pool. Restore the Control Room boundary to OPERABLE status within 24 hours or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

d. With both Control Room Envelope Pressurization Systems inoperable during the performance of Surveillance Requirement 4.7.8.c and the system not being tested under administrative control, immediately suspend the movement of fuel assemblies within the spent fuel pool. Restore at least one inoperable system to OPERABLE status within 4 hours or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6, and fuel movement within containment or the spent fuel pool:

e. With one Control Room Envelope Pressurization System inoperable, restore the inoperable system to OPERABLE status within 7 days. After 7 days, immediately suspend CORE ALTERATIONS and the movement of fuel assemblies.

f. With both Control Room Envelope Pressurization Systems inoperable, immediately suspend CORE ALTERATIONS and the movement of fuel assemblies.

---

* The requirements of Surveillance Requirements 4.7.8.c.2 and 4.7.8.c.3 do not apply during pressure testing of the Cable Spreading Room. This exception is valid until the first entry into MODE 4 following the completion of refueling operations associated with the seventh Refueling Outage.

\# The Control Room boundary may be opened intermittently under administrative control.

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July 24, 2002

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.7.8 Each Control Room Envelope Pressurization System shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the storage air bottles are pressurized to greater than or equal to 2200 psig,

b. At least once per 31 days on a STAGGERED TEST BASIS by verifying that each valve (manual, power operated or automatic) in the flow path not locked, sealed or otherwise secured in position, is in its correct position, and

c. At least once per 24 months or following a major alteration of the control room envelope pressure boundary by:

1. Verifying that the control room envelope is isolated in response to a Control Building Isolation test signal,

2. Verifying that after a 60 second time delay following a Control Building Isolation test signal, the control room envelope pressurizes to greater than or equal to 1/8 inch W.G. relative to adjacent areas and outside atmosphere, and

3. Verifying that the positive pressure of Specification 4.7.8.c.2 is maintained for greater than or equal to 60 minutes.

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PROCEDURES AND PROGRAMS (Continued)

2) Pre-planned operating procedures and backup instrumentation to be used if one or more monitoring instruments become inoperable, and

3) Administrative procedures for returning inoperable instruments to OPERABLE status as soon as practicable.

f. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions*. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 38.57 psig.

The maximum allowable containment leakage rate L_a, at P_a, shall be 0.3 percent by weight of the containment air per 24 hours.

Leakage rate acceptance criteria are:

1) Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq$ ~~0.042~~ 0.06 L_a for all penetrations that are Secondary Containment bypass leakage paths, and $<0.75\ L_a$ for Type A tests;

2) Air lock testing acceptance criteria are:

   a. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   b. For each door, seal leakage rate is $< 0.01\ L_a$ when pressurized to $\geq P_a$.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

* An exemption to Appendix J, Option A, paragraph III.D.2(b)(ii), of 10 CFR Part 50, as approved by the NRC on December 6, 1985.

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATION CHANGES
IMPLEMENTATION OF ALTERNATE SOURCE TERM

RETYPED PAGES OF TECHNICAL SPECIFICATIONS CHANGES

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE containment automatic isolation valve system*, or

   2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are opened under administrative control as permitted by Specification 3.6.3.

b. All equipment hatches are closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of the Containment Leakage Rate Testing Program, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATIONS

1.9 CORE ALTERATIONS shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Federal Guidance No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion."

* In MODE 4, the requirement for an OPERABLE containment isolation valve system is satisfied by use of the containment isolation actuation pushbuttons.

SURVEILLANCE REQUIREMENTS

4.7.7 Each Control Room Emergency Air Filtration System shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 95°F;

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying a system flow rate of 1,120 cfm ±20% and that the system operates for at least 10 continuous hours with the heaters operating;

c. At least once per 24 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Position C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revisions 2, March 1978,* and the system flow rate is 1,120 cfm ± 20%;

   2. Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,* shows the methyl iodide penetration less than or equal to 5.0% when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F), a relative humidity of 70%, and a face velocity of 54 ft/min; and

   3. Verifying a system flow rate of 1,120 cfm ± 20% during system operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,* shows the methyl iodide penetration less than or equal to 5.0% when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F), and a relative humidity of 70%, and a face velocity of 54 ft/min.

e. At least once per 24 months by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.75 inches Water Gauge while operating the system at a flow rate of 1,120 cfm ± 20%;

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PROCEDURES AND PROGRAMS (Continued)

2. Pre-planned operating procedures and backup instrumentation to be used if one or more monitoring instruments become inoperable, and

3. Administrative procedures for returning inoperable instruments to OPERABLE status as soon as practicable.

f. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions*. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 38.57 psig.

The maximum allowable containment leakage rate L_a, at P_a, shall be 0.3 percent by weight of the containment air per 24 hours.

Leakage rate acceptance criteria are:

1. Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.06\ L_a$ for all penetrations that are Secondary Containment bypass leakage paths, and $< 0.75\ L_a$ for Type A tests;

2. Air lock testing acceptance criteria are:

   a. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   b. For each door, seal leakage rate is $< 0.01\ L_a$ when pressurized to $\geq P_a$.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

* An exemption to Appendix J, Option A, paragraph III.D.2(b)(ii), of 10 CFR Part 50, as approved by the NRC on December 6, 1985.

ATTACHMENT 4

PROPOSED TECHNICAL SPECIFICATION CHANGES
IMPLEMENTATION OF ALTERNATE SOURCE TERM

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

Significant Hazards Consideration Determination

This license amendment proposes full implementation of an alternative source term (AST) and changes to the Technical Specifications. Changes are proposed for the following Technical Specifications:

- Definition of Dose Equivalent I-131 - revised to allow use of Federal Guidance Report No. 11 (FGR 11) dose conversion factors
- Technical Specification 3/4.7.7, Control Room Emergency Air Filtration System - changed the value used for methyl iodide penetration test acceptance criteria.
- Technical Specification 3/4.7.8, Control Room Envelope Pressurization System – deleted the specification in its entirety.
- Section 6.8.4.f, the leakage rate acceptance criteria for all penetrations that are secondary containment bypass leakage paths - changed the value used in the acceptance criteria.

We have reviewed the proposed Technical Specifications changes relative to the requirements of 10 CFR 50.92 and determined that a significant hazards consideration is not involved. Specifically, operation of Millstone Power Station Unit 3 with the proposed changes will not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated.*

   The proposed amendment does not involve a significant increase in the probability or consequence of an accident previously analyzed. The Millstone Unit 3 Control Room Emergency Air Filtration System only functions following the initiation of a design basis radiological accident. Therefore, the change to the value used for methyl iodide penetration test acceptance criteria following a design basis accident will not increase the probability of any previously analyzed accident. The Millstone Unit 3 Control Room Envelope Pressurization System is no longer credited in the accident analyses described in the Alternative Source Term (AST) implementation analyses. In accordance with AST implementation analyses, the requirements contained in this Specification do not meet any of 10 CFR 50.36(c)(2)(ii) criteria on items for which Technical Specifications must be established. Deletion of this Technical Specification will not increase the probability of occurence of any previously analyzed accident and does not impact the consequences of any evaluated accident since it is no longer analytically credited. The Millstone Unit 3 containment and the containment systems function to prevent or control the release of radioactive fission products following a postulated accident. Therefore, the change to the value used for the leakage rate acceptance criteria for all penetrations that are secondary containment bypass leakage paths following a design basis accident will not increase the probability of any previously analyzed accident and is limited to ensure it does not increase any accident consequence.

These systems are not initiators of any design bases accident. Revised dose calculations, which take into account the changes proposed by this amendment and the use of the alternative source term, have been performed for the Millstone Unit 3 design basis radiological accidents. The results of these revised calculations indicate that public and control room doses will not exceed the limits specified in 10 CFR 50.67 and Regulatory Guide 1.183. There is not a significant increase in predicted dose consequences for any of the analyzed accidents. Therefore, the proposed changes do not involve a significant increase in the consequences of any previously analyzed accident.

2. *Create the possibility of a new or different kind of accident from any accident previously evaluated.*

The implementation of the proposed changes does not create the possibility of an accident of a different type than was previously evaluated in the UFSAR. Although the proposed changes could affect the operation of the Control Room Emergency Air Filtration System, and containment and the containment systems following a design basis radiological accident, none of these changes can initiate a new or different kind of accident since they are only related to system capabilities that provide protection from accidents that have already occurred. These changes do not alter the nature of events postulated in the UFSAR nor do they introduce any unique precursor mechanisms. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from those previously analyzed.

3. *Involve a significant reduction in the margin of safety.*

The implementation of the proposed changes does not reduce the margin of safety. The proposed changes for the Control Room Emergency Air Filtration System, and containment and the containment systems do not affect the ability of these systems to perform their intended safety functions to maintain dose less than the required limits during design basis radiological events. The revised dose calculations also indicate that the change to the containment depressurization times will continue to maintain the dose to the public and control room operators less than the required limits. The radiological analysis results, when compared with the revised TEDE acceptance criteria, meet the applicable limits. These acceptance criteria have been developed for application to analyses performed with alternative source terms. These acceptance criteria have been developed for the purpose of use in design basis accident analyses such that meeting the stated limits demonstrates adequate protection of public health and safety. It is thus concluded that the margin of safety will not be reduced by the implementation of the changes.

ATTACHMENT 5

PROPOSED TECHNICAL SPECIFICATION CHANGES
IMPLEMENTATION OF ALTERNATE SOURCE TERM

ENVIRONMENTAL IMPACT EVALUATION

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

ENVIRONMENTAL IMPACT EVALUATION

10 CFR51.22(c)(9) provides criteria for and identification of licensing and regulatory action eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not:

1) involve a significant hazards consideration,
2) result in a significant change in the type or a significant increase in the amounts of any effluents that may be released offsite, or
3) result in a significant increase in individual or cumulative occupational exposure.

Dominion Nuclear Conecticut, Inc. (DNC) has reviewed this license amendment and has determined that it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 52.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the proposed license amendment. The basis for this determination is as follows:

1) The proposed license amendment does not involve a significant hazards consideration as described previously in Attachment 4 of this letter.

2) As discussed in the significant hazards evaluation, the changes proposed by this amendment and full implementation of an alternative source term do not result in a significant change or significant increase in the public dose consequences for Millstone Unit 3 design basis radiological accidents. Approval of a new alternative source term for Millstone Unit 3 establishes a new licensing and design basis for assessment of accident consequences. It does not change actual accident sequences; only the regulatory assumptions regarding radiological accidents change. The adoption of an alternative source term, by itself, will not result in plant changes that involve any significant increase in environmental impacts. The proposed changes affect the operation of the Control Room Emergency Air Filtration System during radiological accidents, the requirement for a Control Room Envelope Pressurization System, and the acceptance criteria for the Containment Leakage Testing Program. These systems do not interface with any plant system that is involved in the generation or processing of effluents during normal plant operations. The proposed changes will affect the radioactive effluents during a radiological accident. However, the dose to the public will not exceed the limits specified in 10 CFR 50.67 and Regulatory Guide 1.183. Therefore, implementation of the proposed change and a full alternative source term will not result in a significant change in the types or increase in the amount of any effluents that may be released offsite.

3) The changes proposed by this amendment and full implementation of an alternative source term do not result in a significant increase in control room operator doses during design basis radiological accidents. In addition, the proposed changes do not

require operators or other actions that could increase occupational radiation exposure. The proposed changes will affect the radioactive effluents during a radiological accident. However, the dose to the operator will not exceed the limits specified in 10 CFR 50.67 and GDC-19. Therefore, the proposed amendments and implementation of an alternative source term will not result in a significant increase in individual or cumulative occupational radiation exposure.

ATTACHMENT 6

PROPOSED TECHNICAL SPECIFICATION CHANGES
IMPLEMENTATION OF ALTERNATE SOURCE TERM

MARKED-UP PAGES OF TECHNICAL SPECIFICATION BASES (FOR INFORMATION ONLY)

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guidelines of 10 CFR 50.67 ~~CFR Part 100~~ during accident conditions and the control room operators dose to within the guidelines of ~~GDC 19~~ Regulatory Guide 1.183.

Primary CONTAINMENT INTEGRITY is required in MODES 1 through 4. This requires an OPERABLE containment automatic isolation valve system. In MODES 1, 2 and 3 this is satisfied by the automatic containment isolation signals generated by high containment pressure, low pressurizer pressure and low steamline pressure. In MODE 4 the automatic containment isolation signals generated by high containment pressure, low pressurizer pressure and low steamline pressure are not required to be OPERABLE. Automatic actuation of the containment isolation system in MODE 4 is not required because adequate time is available for plant operators to evaluate plant conditions and respond by manually operating engineered safety features components. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. Since the manual actuation pushbuttons portion of the containment isolation system is required to be OPERABLE in MODE 4, the plant operators can use the manual pushbuttons to rapidly postion all automatic containment isolation valves to the required accident position. Therefore, the containment isolation actuation pushbuttons satisfy the requirement for an OPERABLE containment automatic isolation valve system in MODE 4.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates, as specified in the Containment Leakage Rate Testing Program, ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, Pa. As an added conservatism, the measured overall integrated leakage rate is further limited to less than 0.75 La during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The Limiting Condition for Operation defines the limitations on containment leakage. The leakage rates are verified by surveillance testing as specified in the Containment Leakage Rate Testing Program, in accordance with the requirements of Appendix J. Although the LCO specifies the leakage rates at accident pressure, Pa, it is not feasible to perform a test at such an exact value for pressure. Consequently, the surveillance testing is performed at a pressure greater than or equal to Pa to account for test instrument uncertainties and stabilization changes. This conservative test pressure ensures that the measured leakage rates

3/4.6.6.2 SECONDARY CONTAINMENT

The Secondary Containment is comprised of the containment enclosure building and all contiguous buildings (main steam valve building [partially], engineering safety features building [partially], hydrogen recombiner building [partially], and auxiliary building). The Secondary Containment shall exist when:

a. Each door in each access opening is closed except when the access opening is being used for normal transit entry and exit,

b. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

Secondary Containment ensures that the release of radioactive materials from the primary containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with operation of the Supplementary Leak Collection and Release System, and Auxiliary Building Filter System will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR 50.67 Part 100 during accident conditions.

The SLCRS and the ABF fans and filtration units are located in the auxiliary building. The SLCRS is described in the Millstone Unit No. 3 FSAR, Section 6.2.3.

In order to ensure a negative pressure in all areas within the Secondary Containment under most meteorological conditions, the negative pressure acceptance criterion at the measured location (i.e., 24'6" elevation in the auxiliary building) is 0.4 inches water gauge.

LCO

The Secondary Containment OPERABILITY must be maintained to ensure proper operation of the SLCRS and the auxiliary building filter system and to limit radioactive leakage from the containment to those paths and leakage rates assumed in the accident analyses.

Applicability

Maintaining Secondary Containment OPERABILITY prevents leakage of radioactive material from the Secondary Containment. Radioactive material may enter the Secondary Containment from the containment following a LOCA. Therefore, Secondary Containment is required in MODES 1, 2, 3, and 4 when a design basis accident such as a LOCA could release radioactive material to the containment atmosphere.

3/4.7.1.3 DEMINERALIZED WATER STORAGE TANK (Continued)

If the combined condensate storage tank (CST) and DWST inventory is being credited, there are 50,000 gallons of unusable CST inventory due to tank discharge line location, other physical characteristics, level measurement uncertainty and potential measurement bias error due to the CST nitrogen blanket. To obtain the Surveillance Requirement 4.7.1.3.2's DWST and CST combined volume, this 50,000 gallons of unusable CST inventory has been added to the 334,000 gallon DWST water volume specified in LCO 3.7.1.3 resulting in a 384,000 gallons requirement (334,000 + 50,000 = 384,000 gallons).

3/4.7.1.4 SPECIFIC ACTIVITY

10 CFR 50.67 and Regulatory Guide 1.183

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to ~~a small fraction of 10 CFR Part 100~~ dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm primary-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

January 2, 2003

SURVEILLANCE REQUIREMENTS

For Information Only

For the surveillance requirements, the UHS temperature is measured at the locations described in the LCO write-up provided in this section.

Surveillance Requirement 4.7.5.a verifies that the UHS is capable of providing a 30-day cooling water supply to safety-related equipment without exceeding its design basis temperature. The 24-hour frequency is based on operating experience related to trending of the parameter variations during the applicable modes. This surveillance requirement verifies that the average water temperature of the UHS is less than or equal to 75°F.

Surveillance Requirement 4.7.5.b requires that the UHS temperature be monitored on an increased frequency whenever the UHS temperature is greater than 70°F during the applicable modes. The intent of this Surveillance Requirement is to increase the awareness of plant personnel regarding UHS temperature trends above 70°F. The frequency is based on operating experience related to trending of the parameter variations during the applicable modes.

3/4.7.6 DELETED

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

BACKGROUND

The control room emergency ventilation system provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. Additionally, the system provides temperature control for the control room during normal and post-accident operations.

The control room emergency ventilation system is comprised of the control room emergency air filtration system and a temperature control system.

The control room emergency air filtration system consists of two redundant systems that recirculate and filter the control room air. Each control room emergency air filtration system consists of a moisture separator, electric heater, prefilter, upstream high efficiency particulate air (HEPA) filter, charcoal adsorber, downstream HEPA filter, and fan. Additionally, ductwork, valves or dampers, and instrumentation form part of the system.

Normal Operation

A portion of the control room emergency ventilation system is required to operate during normal operations to ensure the temperature of the control room is maintained at or below 95°F.

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

BACKGROUND (Continued)

Post Accident Operation

The control room emergency ventilation system is required to operate during post-accident operations to ensure the temperature of the control room is maintained and to ensure the control room will remain habitable during and following accident conditions.

The following sequence of events occurs upon receipt of a control building isolation (CBI) signal or a signal indicating high radiation in the air supply duct to the control room envelope.

1. The control room boundary is isolated to prevent outside air from entering the control room to prevent the operators from being exposed to the radiological conditions that may exist outside the control room. The analysis for a loss of coolant accident assumes that the highest releases occur in the first hour after a loss of coolant accident. Delete

2. After 60 seconds, the control room envelope pressurizes to 1/8 inch water gauge by the control room emergency pressurization system. This action provides a continuous purge of the control room envelope and prevents inleakage from the outside environment. Technical Specification 3/4.7.8 provides the requirements for the control room envelope pressurization system.

3. Control room pressurization continues for the first hour.

2 ~~4~~. After one hour, the control room emergency ventilation system will be placed in service in either the 100% recirculation mode (isolated from the outside environment) or filtered pressurization mode (outside air is diverted through the filters to the control room envelope to maintain a positive pressure). The mode of service for the filtration will be based on the radiological conditions that exist outside the control room. To run the control room emergency air filtration system in the filtered pressurization mode, the air supply line must be manually opened.

APPLICABLE SAFETY ANALYSIS

The OPERABILITY of the Control Room Emergency Ventilation System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room. For all postulated design basis accidents except a Fuel Handling Accident, the radiation exposure to personnel occupying the control room shall be 5 rem or less whole body, or its equivalent for the duration of the accident, consistent with the requirements of General Design Criterion 19 of Appendix "A," 10 CFR 50. For a Fuel Handling Accident, the radiation exposure to personnel occupying the control room shall be 5 rem TEDE or less, consistent with the requirements of 10 CFR 50.67. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50. Deleted

For Information Only

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

LIMITING CONDITION FOR OPERATION

Two independent control room emergency air filtration systems are required to be operable to ensure that at least one is available in the event the other system is disabled.

A control room emergency air filtration system is OPERABLE when the associated:

a. Fan is OPERABLE;

b. HEPA filters and charcoal adsorbers are not excessively restricting flow and are capable of performing their filtration functions; and

c. moisture separator, heater, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

The integrity of the control room habitability boundary (i.e., walls, floors, ceilings, ductwork, and access doors) must be maintained such that the control building habitability zone can be maintained at its design positive pressure if required to be aligned in the filtration pressurization mode. However, the LCO is modified by a footnote allowing the control room boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in constant communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, 4, 5, and 6.
During fuel movement within containment or the spent fuel pool.

Actions a., b., and c. of this specification are applicable at all times during plant operation in MODES 1, 2, 3, and 4. Actions d. and e. are applicable in MODES 5 and 6, and whenever fuel is being moved within containment or the spent fuel pool. The fuel handling accident analyses assume that during a fuel handling accident some of the fuel that is dropped and some of the fuel impacted upon is damaged. Therefore, the movement of either new or irradiated fuel (assemblies or individual fuel rods) can cause a fuel handling accident, and this specification is applicable whenever new or irradiated fuel is moved within the containment or the storage pool.

For Information Only

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

ACTIONS

Modes 1, 2, 3, and 4

a. With one control room emergency air filtration system inoperable, action must be taken to restore the inoperable system to an OPERABLE status within 7 days. In this condition, the remaining control room emergency air filtration system is adequate to perform the control room protection function. However, the overall reliability is reduced because a single failure in the OPERABLE train could result in a loss of the control room emergency air filtration system function. The 7-day completion time is based on the low probability of a DBA occurring during this time period, and the ability of the remaining train to provide the required capability.

If the inoperable train cannot be restored to an OPERABLE status within 7 days, the unit must be placed in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. These completion times are reasonable, based on operating experience, to reach the required unit condition from full power conditions in an orderly manner and without challenging unit systems.

b. With both control room emergency air filtration systems inoperable, except due to an inoperable control room boundary, the movement of fuel within the spent fuel pool must be immediately suspended. At least one control room emergency air filtration system must be restored to OPERABLE status within 1 hour, or the unit must be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. These completion times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

c. With both control room emergency air filtration systems inoperable due to an inoperable control room boundary, the movement of fuel within the spent fuel pool must be immediately suspended. The control room boundary must be restored to OPERABLE status within 24 hours, or the unit must be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

If the control room boundary is inoperable in MODES 1, 2, 3, and 4, the control room emergency air filtration systems cannot perform their intended functions. Actions must be taken to restore on OPERABLE control room boundary within 24 hours. During the period that the control room boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be

For Information Only

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

ACTIONS (Continued)

available to address these concerns for intentional and unintentional entry in to this condition. The 24 hour allowed outage time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour allowed outage time is a typically reasonable time to diagnose, plan, and possibly repair, and test most problems with the control room boundary.

MODES 5 and 6, and fuel movement within containment or the spent fuel pool

d. With one control room emergency air filtration system inoperable, action must be taken to restore the inoperable system to an OPERABLE status within 7 days. After 7 days, either initiate and maintain operation of the remaining OPERABLE control room emergency air filtration system in the recirculation mode or suspend the movement of fuel. Initiating and maintaining operation of the OPERABLE train in the recirculation mode ensures: (i) operability of the train will not be compromised by a failure of the automatic actuation logic; and (ii) active failures will be readily detected.

e. With both control room emergency air filtration systems inoperable, or with the train required by ACTION 'd' not capable of being powered by an OPERABLE emergency power source, actions must be taken to suspend all operations involving the movement of fuel. This action places the unit in a condition that minimizes risk. This action does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

4.7.7.a

The control room environment should be checked periodically to ensure that the control room temperature control system is functioning properly. Verifying that the control room air temperature is less than or equal to 95°F at least once per 12 hours is sufficient. It is not necessary to cycle the control room ventilation chillers. The control room is manned during operations covered by the technical specifications. Typically, temperature aberrations will be readily apparent.

4.7.7.b

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing the trains once every 31 days on a STAGGERED TEST BASIS provides an adequate check of this system. This surveillance requirement verifies a system flow rate of 1,120 cfm ± 20%. Additionally, the system is required to operate for at least 10 continuous hours with the heaters energized. These operations are sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters due to the humidity in the ambient air.

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

SURVEILLANCE REQUIREMENTS (Continued) For Information Only

4.7.7.c

The performance of the control room emergency filtration systems should be checked periodically by verifying the HEPA filter efficiency, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. The frequency is at least once per 24 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system.

ANSI N510-1980 will be used as a procedural guide for surveillance testing.

4.7.7.c.1

This surveillance verifies that the system satisfies the in-place penetration and bypass leakage testing acceptance criterion of less than 0.05% in accordance with Regulatory Position C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, while operating the system at a flow rate of 1,120 cfm ± 20%. ANSI N510-1980 is used in lieu of ANSI N510-1975 referenced in the regulatory guide.

4.7.7.c.2

This surveillance requires that a representative carbon sample be obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978 and that a laboratory analysis verify that the representative carbon sample meets the laboratory testing criteria of ASTM D3803-89 and Millstone Unit 3 specific parameters. The laboratory analysis is required to be performed within 31 days after removal of the sample. ANSI N510-1980 is used in lieu of ANSI N510-1975 referenced in Revision 2 of Regulatory Guide 1.52.

4.7.7.c.3

This surveillance verifies that a system flow rate of 1,120 cfm ± 20%, during system operation when testing in accordance with ANSI N510-1980.

4.7.7.d

After 720 hours of charcoal adsorber operation, a representative carbon sample must be obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and a laboratory analysis must verify that the representative carbon sample meets the laboratory testing criteria of ASTM D3803-89 and Millstone Unit 3 specific parameters.

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July 24, 2002

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

The laboratory analysis is required to be performed within 31 days after removal of the sample. ANSI N510-1980 is used in lieu of ANSI N510-1975 referenced in Revision 2 of Regulatory Guide 1.52.

The maximum surveillance interval is 900 hours, per Surveillance Requirement 4.0.2. The 720 hours of operation requirement originates from Nuclear Regulatory Guide 1.52, Table 2, Note C. This testing ensures that the charcoal adsorbency capacity has not degraded below acceptable limits as well as providing trending data.

4.7.7.e.1

This surveillance verifies that the pressure drop across the combined HEPA filters and charcoal adsorbers banks at less than 6.75 inches water gauge when the system is operated at a flow rate of 1,120 cfm ± 20%. The frequency is at least once per 24 months.

4.7.7.e.2

This surveillance verifies 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. The frequency is at least once per 24 months.

The intent of this surveillance is to verify the ability of the control room emergency air filtration system to maintain a positive pressure while running in the filtered pressurization mode.

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

Deleted

During the first hour, the control room pressurization system creates and maintains the positive pressure in the control room. This capability is verified by Surveillance Requirement 4.7.8.C, independent of Surveillance Requirement 4.7.7.e.2. A CBI signal will automatically align an operating filtration system into the recirculation mode of operation due to the isolation of the air supply line to the filter.

After the first hour of an event with the potential for a radiological release, the control room emergency ventilation system will be aligned in either the recirculation mode (isolated from the outside environment) or filtered pressurization mode (outside air is diverted through the filters to the control room envelope to maintain a positive pressure). The mode of service for the control room emergency air filtration system will be based on the radiological conditions that exist outside the control room. Alignment to the filtered pressurization mode requires manual operator action to open the air supply line.

4.7.7.e.3

This surveillance verifies that the heaters can dissipate 9.4 ± 1 kW at 480V when tested in accordance with ANSI N510-1980. The frequency is at least once per 24 months. The heater kW measured must be corrected to its nameplate rating. Variations in system voltage can lead to measurements of kW which cannot be compared to the nameplate rating because the output kW is proportional to the square of the voltage.

4.7.7.f

Following the complete or partial replacement of a HEPA filter bank, the operability of the cleanup system should be confirmed. This is accomplished by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criterion of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 1,120 cfm ± 20%.

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

4.7.7.q

Following the complete or partial replacement of a charcoal adsorber bank, the operability of the cleanup system should be confirmed. This is accomplished by verifying that the cleanup system satisfied the in-place penetration and bypass leakage testing acceptance criterion of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow of 1,120 cfm ± 20%.

References:

(1) Nuclear Regulatory Guide 1.52, Revision 2
(2) MP3 UFSAR, Table 1.8-1, NRC Regulatory Guide 1.52
(3) NRC Generic Letter 91-04
(4) Condition Report (CR) #M3-99-0271

3/4.7.8 ~~CONTROL ROOM ENVELOPE PRESSURIZATION SYSTEM~~ DELETED

~~BACKGROUND~~

~~The control room envelope pressurization system provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity.~~

~~The control room envelope pressurization system consists of two banks of air bottles with its associated piping, instrumentation, and controls. Each bank is capable of providing the control room area with one-hour of air following any event with the potential for radioactive releases.~~

~~Control Room Envelope OPERABILITY is satisfied while:~~

- ~~Door 352 (C-49-1) is closed (East door)~~
- ~~Door 351 (C-47-1) is closed, but C-47-1A, ATD/Missile Shield, is not closed (West doors)~~

~~Normal Operation~~

~~During normal operations, the control room envelope pressurization system is required to be on standby.~~

~~Post Accident Operation~~

~~The control room envelope pressurization system is required to operate during post-accident operations to ensure the control room will remain habitable during and following accident conditions.~~

~~The sequence of events which occurs upon receipt of a control building isolation (CBI) signal or a signal indicating high radiation in the air supply duct to the control room envelope is described in Bases Section 3/4.7.7.~~

3/4.7.8 CONTROL ROOM ENVELOPE PRESSURIZATION SYSTEM (Continued)

APPLICABLE SAFETY ANALYSIS

The OPERABILITY of the control room envelope pressurization system ensures that: (1) breathable air is supplied to the control room, instrumentation rack room, and computer room, and (2) a positive pressure is created and maintained within the control room envelope during control building isolation for the first hour following any event with the potential for radioactive releases. Each system is capable of providing an adequate air supply to the control room for one hour following an initiation of a control building isolation signal. After one hour, operation of the control room emergency ventilation system would be initiated.

LIMITING CONDITION FOR OPERATION

Two independent control room envelope pressurization systems are required to be operable to ensure that at least one is available in the event the other system is disabled.

A control room envelope pressurization system is OPERABLE when the associated:

a. air storage bottles are OPERABLE; and

b. piping and valves are OPERABLE.

The integrity of the control room habitability boundary (i.e., walls, floors, ceilings, ductwork, and access doors) must be maintained. However, the LCO is modified by a footnote allowing the control room boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in constant communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated.

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3/4.7.8 CONTROL ROOM ENVELOPE PRESSURIZATION SYSTEM (Continued)

APPLICABILITY

In MODES 1, 2, 3, 4, 5, and 6.
During fuel movement within containment or the spent fuel pool.

Actions a., b., c., and d. of this specification are applicable at all times during plant operation in MODES 1, 2, 3, and 4. Actions e. and f. are applicable in MODES 5 and 6, and whenever fuel is being moved within containment or the spent fuel pool. The fuel handling accident analyses assume that during a fuel handling accident some of the fuel that is dropped and some of the fuel that is impacted upon is damaged. Therefore, the movement of either new or irradiated fuel (assemblies or individual fuel rods) can cause a fuel handling accident, and this specification is applicable whenever new or irradiated fuel is moved within the containment or the storage pool.

ACTIONS

MODES 1, 2, 3, and 4

a. With one control room envelope pressurization system inoperable, action must be taken either to restore the inoperable system to an OPERABLE status within 7 days, or place the unit in HOT STANDBY within six hours and COLD SHUTDOWN within the next 30 hours.

The remaining control room envelope pressurization system is adequate to perform the control room protection function. However, the overall reliability is reduced because a single failure in the OPERABLE train could result in a loss of the control room envelope pressurization system. The 7-day completion time is based on the low probability of a design basis accident occurring during this time period and the ability of the remaining train to provide the required capability.

The completion times for the unit to be placed in HOT STANDBY and COLD SHUTDOWN are reasonable. They are based on operating experience, and they permit the unit to be placed in the required conditions from full power conditions in an orderly manner and without challenging unit systems.

b. With both control room envelope pressurization systems inoperable, except due to an inoperable control room boundary or during performance of Surveillance Requirement 4.7.8.c, the movement of fuel within the spent fuel pool must be immediately suspended. At least one control room envelope pressurization system must be restored to OPERABLE status within 1 hour, or the unit must be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. These completion times are reasonable, based in operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

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3/4.7.8 CONTROL ROOM ENVELOPE PRESSURIZATION SYSTEM (Continued)

ACTIONS (Continued)

c. With both control room envelope pressurization systems inoperable due to an inoperable control room boundary, the movement of fuel within the spent fuel pool must be immediately suspended. The control room boundary must be restored to OPERABLE status within 24 hours, or the unit must be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

If the control room boundary is inoperable in MODES 1, 2, 3, and 4, the control room envelope pressurization systems cannot perform their intended functions. Actions must be taken to restore an OPERABLE control room boundary within 24 hours. During the period that the control room boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry in to this condition. The 24 hour allowed outage time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour allowed outage time is a typically reasonable time to diagnose, plan, and possibly repair, and test most problems with the control room boundary.

d. With both control room envelope pressurization systems inoperable during the performance of Surveillance Requirement 4.7.8.c and the system not being tested under administrative control, the movement of fuel within the spent fuel pool must be immediately suspended. At least one control room envelope pressurization system must be restored to OPERABLE status within 4 hours, or the unit must be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The administrative controls for the system not being tested consist of a dedicated operator, in constant communication with the control room, who can rapidly restore this system to OPERABLE status. Allowing both control room envelope pressurization systems to be inoperable for 4 hours under administrative control is acceptable since the system not being tested is inoperable only because it is isolated. Therefore, the system can be rapidly restored if needed. The other completion times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

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3/4.7.8 CONTROL ROOM ENVELOPE PRESSURIZATION SYSTEM (Continued)

ACTIONS (Continued)

MODES 5 and 6, and fuel movement within containment or the spent fuel pool

e. With one control room envelope pressurization system inoperable, action must be taken to restore the inoperable system to an OPERABLE status within 7 days. After 7 days, immediately suspend the movement of fuel. This action places the unit in a condition that minimizes potential radiological exposure to Control Room personnel. This action does not preclude the movement of fuel to a safe position.

The remaining control room envelope pressurization system is adequate to perform the control room protection function. However, the overall reliability is reduced because a single failure in the OPERABLE train could result in a loss of the control room envelope pressurization system. The 7-day completion time is based on the low probability of a design basis accident occurring during this time period and the ability of the remaining train to provide the required capability.

Stud tensioning may continue in MODE 6 and a MODE change to MODE 5 is permitted with a control room envelope pressurization system inoperable (Reference 1).

f. With both control room envelope pressurization systems inoperable, immediately suspend the movement of fuel. This action places the unit in a condition that minimizes potential radiological exposure to Control Room personnel. This action does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

4.7.8.a

This surveillance requires verification that the air bottles are properly pressurized. Verifying that the air bottles are pressurized to greater than or equal to 2200 psig will ensure that a control room envelope pressurization system will be capable of supplying the required flow rate. The frequency of the surveillance is at least once per 7 days. It is based on engineering judgment and has been shown to be appropriate through operating experience.

4.7.8.b

This surveillance requires verification of the correct position of each valve (manual, power operated, or automatic) in the control room envelope pressurization system flow path. It helps ensure that the control room envelope pressurization system is capable of performing its intended safety function by verifying that an appropriate flow path will exist. The surveillance applies to those valves that could be mispositioned. This surveillance does not apply to valves that have been locked, sealed, or secured in position, because these positions are verified prior to locking, sealing, or securing.

The frequency of the surveillance is at least once per 31 days on a STAGGERED TEST BASIS. It is based on engineering judgment and has been shown to be appropriate through operating experience.

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3/4.7.8 CONTROL ROOM ENVELOPE PRESSURIZATION SYSTEM (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

4.7.8.c

The performance of the control room envelope pressurization system should be checked periodically. The frequency is at least once per 24 months and following any major alteration of the control room envelope pressure boundary.

A major alteration is a change to the control room envelope pressure boundary that: (1) results in a breach greater than analyzed for acceptable pressurization and requires nonroutine work evolutions to restore the boundary. A nonroutine work evolution is one which makes it difficult to determine As-Found and As-Left conditions. Examples of routine work evolution include: (1) opening and closing a door, and (2) repairing cable and pipe penetrations because the repairs are conducted in accordance with procedures and are verified via inspections. For these two examples, there is a high level of assurance that the boundary is restored to the As-Found condition.

This surveillance requires at least once per 24 months or following a major alteration of the control room envelope pressure boundary by:

- Verifying the control room envelope is isolated in response to a Control Building Isolation Test signal,
- Verifying, after a 60 second time delay following a Control Building Isolation Test signal, the control room envelope pressurizes to greater than or equal to 0.125 inch water gauge relative to adjacent areas and outside atmosphere; and
- Verifying the positive pressure of Technical Specification 4.7.8.c.2 is maintained for greater than or equal to 60 minutes.

Changes in conditions outside the control room envelope cause pressure spikes which are reflected on the differential pressure indicator, 3HVC-PDI 113.

Pressure spikes or fluctuations which result in the differential pressure momentarily dropped below the 0.125 inch water gauge acceptance criteria are acceptable providing the following conditions are met:

1. Differential pressure remains positive at all times.
2. Differential pressure is only transitorily below the acceptance criteria.
3. Differential pressure returns to a value above the acceptance criteria.

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3/4.7.8 CONTROL ROOM ENVELOPE PRESSURIZATION SYSTEM (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

The control room envelope pressurization system design basis criteria is set at ≥ 0.125 inch water gauge criteria to account for wind effects, thermal column effects, and barometric pressure changes. Pressurizing the control room envelope of 0.125 inch water gauge above the initial atmospheric pressure ensures it will remain at a positive pressure during subsequent changes in outside conditions over the next 60 minutes. Since the surveillance requirement is verified by actual reference to outside pressure, allowances are provided for differential pressure fluctuations caused by external forces. The 0.125 inch water gauge acceptance criteria provides the margin for these fluctuations. This meets the requirements of Regulatory Guide 1.78 and NUREG-800, Section 6.4 and is consistent with the assumptions in the Control Room Operator DBA dose calculation.

4.7.8.c.1

This surveillance verifies that the control room envelope is isolated following a control building isolation (CBI) test signal.

4.7.8.c.2

This surveillance verifies that the control room envelope pressurizes to greater than or equal to 1/8 inch water gauge, relative to the outside atmosphere, after 60 seconds following receipt of a CBI test signal.

4.7.8.c.3

This surveillance verifies that the positive pressure developed in accordance with Surveillance Requirement 4.7.8.c.2 is maintained for greater than or equal to 60 minutes. This capability is independent from the requirements regarding the control room emergency filtration system contained in Technical Specification 3/4.7.7. Also, following the first hour, the control room emergency ventilation system is responsible for ensuring that the control room envelope remains habitable.

References:

(1) NRC Routine Inspection Report 50-423/87-33, dated February 10, 1988.

(2) NRC Generic Letter 91-04.

BASES

3/4.9.10 AND 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND STORAGE POOL

at least

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.