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June 13, 2006

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
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Serial No.	05-662
MPS Lic/MAE	R0
Docket No.	50-336
License No.	DPR-65

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2
LICENSE BASIS DOCUMENT CHANGE REQUEST (LBDCR) 04-MP2-011
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 DPR-65 for Millstone Power Station Unit 2. 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.

DNC has evaluated the proposed technical specifications changes and has determined that they do not involve a significant hazards consideration as defined in 10 CFR 50.92. The basis for DNC's determination is provided in Attachment 4. DNC has also determined that 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 this determination is provided in Attachment 5.

The Site Operations Review Committee has reviewed and concurred with the determinations.

DNC requests approval and issuance of this amendment with the amendment to be implemented within 90 days of issuance.

A001

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)

1. Discussion of Changes
2. Marked Up Technical Specification Pages
3. Proposed Technical Specification Pages
4. Significant Hazards Consideration
5. Environmental Evaluation
6. Marked Up Associated Bases Pages (For Information Only)

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 13th day of June, 2006.

My Commission Expires: August 31, 2008.

Margaret B. Bennett
NOTARY PUBLIC

(SEAL)

Serial No. 05-662
Docket No. 50-336

ATTACHMENT 1

LICENSE BASIS DOCUMENT CHANGE REQUEST (LBDCR) 04-MP2-011
PROPOSED TECHNICAL SPECIFICATION CHANGES
IMPLEMENTATION OF ALTERNATE SOURCE TERM

DISCUSSION OF CHANGES

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

LICENSE BASIS DOCUMENT CHANGE REQUEST (LBDCR) 04-MP2-011
PROPOSED TECHNICAL SPECIFICATION CHANGES
IMPLEMENTATION OF ALTERNATE SOURCE TERM

DISCUSSION OF CHANGES

Table of Contents

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), "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," (Reference 1) accident methodology for Millstone Power Station Unit 2 (MPS2). The accident source term discussed in Reference 1 is herein referred to as the alternate 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 alternate source terms. The results have been compared with the acceptance criteria contained either in 10 CFR 50.67, "Accident Source Term," (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 MPS2.
- Allow MPS2 to achieve a consistent design basis for all accident dose assessments.
- Increase operational flexibility by allowing increased unfiltered control room in-leakage.
- Increase operational flexibility by allowing increased secondary containment bypass leakage.
- Increase operational flexibility by allowing increased refueling water storage tank (RWST) backleakage.
- Increase the primary to secondary leak rate limit.
- Redefine the Dose Equivalent I-131 definition.
- Approve an operator action to isolate the control room within 4 hours following a main steam line break.
- Allow the time for an operator to place the control room emergency ventilation system (CREV) in service to increase following a control room isolation.
- Clarify existing technical specifications.

The radiological dose analyses discussed in this report were performed with a controlled version of the computer code RADTRAD-NAI 1.1a (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 10 CFR 50, Appendix B.

Control room atmospheric dispersion factors were evaluated using the ARCON96 computer code (Reference 5). The ORIGENS 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 MPS2 Updated Final Safety Analysis Report (UFSAR) consist of assessments of the following events:

1) Main Steam Line Break (MSLB)
2) Control Rod Ejection Accident (CREA)
3) Steam Generator Tube Rupture (SGTR)
4) Loss of Coolant Accident (LOCA)
5) Fuel Handling Accident (FHA)
6) Spent Fuel Cask Drop Accident*
7) Waste Gas System Failure (WGSF)

*The MPS2 Spent Fuel Cask Drop Accident is a subset of accidents represented as a Fuel Handling Accident in FSAR Section 14.7.4. It is an event in the spent fuel pool where a shielded cask tips over and damages all fuel within the potential impact area.

1.3 Analysis Assumptions & Key Parameter Values

1.3.1 Selection of Events Requiring Reanalysis

Standard Review Plan (SRP) 15.0.1, Section 1, Item Number 4, supersedes the radiological analyses assumptions, acceptance criteria, and methodologies identified in the following SRP sections for PWRs:

a. Section 15.1.5, Steam System Piping Failures Inside and Outside of Containment.

b. Sections 15.3.3-15.3.4, Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break.

c. Section 15.4.8, Spectrum of Rod Ejection Accidents.

d. Section 15.6.2, Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment.

e. Section 15.6.3, Radiological Consequences of Steam Generator Tube Failure.

f. Section 15.6.5, Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant System Pressure Boundary.

g. Section 15.7.4, 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 MPS2. To support the licensing and plant operation changes discussed in Section 2.0 of this application, the loss of coolant accident (LOCA) and fuel handling accident (FHA) were reanalyzed employing the RG 1.183 source term. To ensure that all accident analyses have a consistent basis, the following accidents were also reanalyzed:

- Main Steam Line Break (MSLB) Accident;
- Control Rod Ejection Accident (CREA); 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 MPS2.

The Waste Gas System Failure radiological analysis is being relocated to FSAR Chapter 11, Radioactive Waste Processing and Radiation Protection Systems. This event is unaffected by the conversion to AST. Whole body dose and thyroid

dose will be retained with the acceptance criteria remaining as "substantially below 10 CFR Part 100 guidelines".

The proposed licensing and plant operational changes are discussed in Section 2.0. These changes require appropriate changes to the MPS2 operating license and technical specifications, which are also described in Section 2.0 of this report. The changes considered as a result of the re-analyses and clarifications are listed below:

a. Revise definition of Dose Equivalent I-131 in Section 1.19 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," 1988, as the source of inhalation thyroid dose conversion factors (Reference 8).

b. Change the combined leakage rate acceptance criteria for all penetrations that are secondary containment bypass leakage paths in Technical Specification LCO 3.6.1.2, "Containment Leakage" from $< 0.0072\ L_a$ to $< 0.014\ L_a$

c. Change the control room unfiltered in-leakage limit in Technical Specification SR 4.7.6.1.e.3 from 130 SCFM to 200 SCFM and clarify the applicability.

d. Revise the primary to secondary leakage limit in Technical Specification LCO 3.4.6.2.c from 0.035 GPM to 75 GPD through any one steam generator.

e. Approve a new operator action to isolate the control room up to 4 hours after a MSLB.

f. Approve a timing change to an existing operator action such that within 1 hour (instead of 10 minutes) of isolation, the control room emergency ventilation system is placed in filtered recirculation.

g. Delete Technical Specification LCO 3.3.4 and SR 3.3.4, "Containment Purge Valve Isolation Signal," since its intent is addressed by LCO 3.6.3.2, "Containment Ventilation System."

h. Revise the language of Technical Specification LCO 3.3.3.1 Action b to be consistent with similar action statements within that technical specification.

i. Revise the technical specification bases to reflect the above listed changes, as appropriate, in accordance with Technical Specification 6.23, "Bases Control Program."

These changes require reanalysis of the LOCA, FHA, SGTR, MSLB, and CREA. Sections 3.1 through 3.5, 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 3.5 of this attachment provide specific assumptions that were employed for the LOCA, FHA, SGTR, MSLB and CREA, respectively.

The dose analyses documented in this application employ the TEDE calculation method as specified in RG 1.183 for AST applications. 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 MPS2 control room personnel are calculated for the assumed 30-day duration of the event (2-hour duration for LPZ for the fuel handling accident).

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 common analysis assumptions and plant features that are used in the analysis of all of the events. These items are presented in Tables 1.3-1 through 1.3-5.

Table 1.3-1

General Control Room Assumptions & Key Parameters Employed in the AST Analyses

Assumption / Parameter	Value
Control Room Effective Volume	35,656 ft3
Normal Control Room Intake Flow Rate prior to Isolation	800 cfm
Unfiltered In-leakage	200 cfm
Emergency Ventilation System Recirculation Flow Rate	2250 cfm
Response Time for Control Room to Isolate upon Receipt of Control Room Ventilation Radiation Monitor Alarm Signal *(Note: this value is validated for each accident analysis in the referenced calculations where the alarm signal is generated by the radiation monitor)*	20 seconds
Response Time for Control Room to Isolate upon Receipt of Enclosure Building Filtration Actuation/ SI signal	20 seconds
Time credited for operator action to align Control Room Emergency Ventilation System after isolation to filtered, recirculation mode	1 hour
Filter Efficiencies*	≤ 90% elemental ≤ 90% aerosol ≤ 70% organic
Millstone Unit 2 Control Building Wall Thickness:	2 feet Concrete
Millstone Unit 2 Control Room Ceiling/Roof Thickness:	2 feet Concrete
Minimum Distance Between Control Room Filters and Control Room Operators	24.5 feet (includes 2 feet thick concrete)
Millstone Unit 2 Control Room Occupancy Factors	
0 – 24 hours	1.0
24 – 96 hours	0.6
96 – 720 hours	0.4

* ≤ 90% efficiency for elemental/ aerosol and 70% for organic are conservatively used in the analysis. Actual design basis is 90% for all iodine species.

Table 1.3-2
Containment Parameters Commonly Employed in the AST Analyses

Assumption / Parameter	Value
Containment Free Volume	1.899E6 ft^3
Millstone Unit 2 Containment Wall Thickness:	3 ft 9 in of Concrete
Millstone Unit 2 Containment Dome Thickness:	3 ft 3 in of Concrete
Distance from Millstone Unit 2 Containment to the Millstone Unit 2 Control Room for containment shine dose determination:	distance not credited – assumption of contact with containment used
Millstone Unit 2 Containment Inner Radius:	65 ft

Table 1.3-3
Offsite Atmospheric Dispersion Factors (sec/m3)*

Receptor/ Source Location / Duration	X/Q (sec/m^3)
Exclusion Area Boundary (EAB)	
Millstone Stack (includes fumigation)	1.00E-04
All other release points (ground level)	3.66E-04
Low Population Zone (LPZ)	
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
All other release points (ground level)	
0 – 4 hours	4.80E-05
4 – 8 hours	2.31E-05
8 – 24 hours	1.60E-05
24 – 96 hours	7.25E-06
96 – 720 hours	2.32E-06

* Unchanged from existing, approved values

Table 1.3-4
Control Room Atmospheric Dispersion Factors

Source Location / Duration		X/Q (sec/m^3)
Millstone Stack *		
	0 – 4 hour	2.51E-04
	4 – 8 hour	1.96E-05
	8 – 24 hour	5.46E-06
	24 – 96 hour	3.43E-07
	96 – 720 hour	6.44E-09
Containment Enclosure Building- Ground Release**		
	0 – 2 hour	3.00E-03
	2 – 8 hour	1.87E-03
	8 – 24 hour	6.64E-04
	24 – 96 hour	5.83E-04
	96 – 720 hour	4.97E-04
Refueling Water Storage Tank (RWST) Vent**		
	0 – 2 hour	9.54E-04
	2 – 8 hour	7.56E-04
	8 – 24 hour	2.72E-04
	24 – 96 hour	2.17E-04
	96 – 720 hour	1.51E-04
Atmospheric Dump Valves (ADVs)** & Enclosure Building Blowout Panels***		
	0 – 2 hour	7.40E-03
	2 – 8 hour	5.71E-03
	8 – 24 hour	2.13E-03
	24 – 96 hour	1.74E-03
	96 – 720 hour	1.43E-03
Turbine Building Blowout Panels**		
	0 – 2 hour	1.22E-02
	2 – 8 hour	8.67E-03
	8 – 24 hour	3.77E-03
	24 – 96 hour	2.92E-03
	96 – 720 hour	2.23E-03
Main Steam Safety Valves (MSSVs)**		
	0 – 2 hour	3.03E-03
	2 – 8 hour	2.30E-03
	8 – 24 hour	8.46E-04
	24 – 96 hour	6.73E-04
	96 – 720 hour	5.49E-04

* - previously approved in Reference 13

** - Enclosure 1 of Attachment 1 provides details on specific ARCON96 modeling. Supporting meteorological data and documentation provided via Reference 11.

*** - X/Qs from the Enclosure Building blowout panel closest to the control room are conservatively assumed to be the same as those for the ADV because the ADV is closer to the control room than the blowout panel and the ADV is a point source versus a large area source for the blowout panel.

Table 1.3-5
Breathing Rates

Source Location / Duration		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 based on the MPS2 AST analyses accompanying this license amendment request.

2.1 Implementation of RG 1.183 Methods as Design Basis Source Term

This report supports a request to revise the design basis accident source term for MPS2. 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 10 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 under 10 CFR 50.90.

2.2 Increase in the Control Room Unfiltered In-leakage Rate

Control room unfiltered in-leakage testing is performed to ensure that the in-leakage does not exceed the criteria specified in the technical specifications. The technical specification acceptance criteria were developed to ensure acceptable dose consequences to the control room operators following a radiological event. The proposed change in the unfiltered in-leakage criteria, from 130 SCFM to 200 SCFM, 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 10 CFR 50.67 and RG 1.183. This impacts Technical Specification SR 4.7.6.1.e.3.

In addition, clarification will be made regarding the APPLICABILITY of Technical Specification LCO 3.7.6.1. ACTIONs 3.7.6.1.d and 3.7.6.1.e are stated as applicable during "MODES 5 and 6 and during irradiated fuel movement within containment or the spent fuel pool." This statement will be modified to "MODES 5 and 6 <u>or</u> during irradiated fuel movement within containment or the spent fuel pool."

2.3 Increase in the Secondary Containment Bypass Leakage Rate

Secondary containment bypass leakage testing is performed to ensure that the leakage rate does not exceed the leakage rate values as specified in the technical specifications. The technical specification acceptance criteria were

developed to ensure acceptable dose consequences following a radiological event. The proposed increase in the acceptable secondary containment bypass leakage rate from < 0.0072 L_a to < 0.014 L_a, 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 10 CFR 50.67 and RG 1.183. This impacts Technical Specification LCO 3.6.1.2.c.

2.4 Increase in the RCS Primary-to-Secondary Leak Rate

Technical specification limits on RCS primary-to-secondary leakage are necessary to ensure that dose consequences to the general public and the control room operators following a radiological event fall within the acceptance criteria of 10 CFR 50.67 and RG 1.183. The proposed increase in the primary-to-secondary leak rate limit from 0.035 GPM (equivalent to 50.4 GPD) to 75 GPD, 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 10 CFR 50.67 and RG 1.183. It should be noted that the proposed radiological design basis analyses are based on a primary-to-secondary leak rate of 150 GPD while 75 GPD is the proposed leak rate limit in the technical specifications. The current design basis analyses limit is 0.035 GPM (50.4 GPD). The higher leak rate assumed in the design analyses provides a conservative margin beyond the proposed technical specification requirements in consideration of Generic Letter 2006-01, "Steam Generator Tube Integrity and Associated Technical Specifications," and the associated TSTF-449. This impacts Technical Specification LCO 3.4.6.2.c.

2.5 Approve a New Operator Action

A new operator action is required to isolate the control room within 4 hours following a MSLB and initiate filtered recirculation consistent with Section 2.6. This ensures that dose consequences to the control room operators are within the acceptance criteria of 10 CFR 50.67.

2.6 Approve a Change to an Existing Operator Action

A change to an existing operator action is required to reduce operator burden. The current operator action requires that following control room isolation, control room emergency ventilation must be aligned to the filtered recirculation mode within 10 minutes. The proposed accident dose evaluations are assessed assuming CREV is aligned at 1 hour instead of 10 minutes to minimize prompt steps that an operator must perform following an accident.

2.7 Redefine Dose Equivalent I-131

Dose equivalent I-131 (DEQ I-131) is defined in the technical specifications for the purpose of assessing RCS specific activity relative to operating limits as well as for specific reporting requirements. The proposed change ensures that the technical specification definition is consistent with the analysis of the consequences from radiological events using the coolant source term at the technical specification DEQ I-131 specific activity limit specified by RG 1.183. This impacts Technical Specification Definition 1.19.

2.8 Control Room X/Qs

MPS2 control room X/Qs have been created or revised to reflect the requirements of RG 1.194 (Reference 10).

2.9 Deletion of TS 3/4.3.4

Technical Specification LCO 3.3.4 and all associated surveillance requirements and bases are being eliminated. This LCO requires operability of a containment purge valve isolation signal during MODES 1 through 4. Since LCO 3.6.3.2 requires that the purge valves be sealed and locked closed during Modes 1 through 4, there is no reason to maintain operability of the signal while in Modes 1 through 4. No credit is taken in any radiological accident analysis for operability of the signal. Since the system is not credited during Modes 1 through 4, the elimination of the technical specification is proposed. Removal of the technical specification is further warranted since it does not meet the criteria in 10 CFR 50.36 for inclusion into the technical specifications.

2.10 Clarification of TS 3/4.3.3

Technical Specification LCO 3.3.3.1, Action b, contains wording that is inconsistent with ACTION descriptions for Table 3.3-6. This inconsistent wording has been found to be confusing.

Action b states "With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6..." Action b is being rewritten, consistent with other action statements in this technical specification, to say "With the number of operable channels less than required by the MINIMUM CHANNELS OPERABLE requirements, take the ACTION shown in Table 3.3-6..."

2.11 Miscellaneous Bases Only Changes

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

Clarification will be provided regarding timing of operator actions and the time requirements to perform those actions to isolate the control room and subsequent alignment of control room emergency ventilation to the filtered, recirculation mode. In addition, clarification will be made with regard to APPLICABILITY of certain ACTION statements for control room emergency ventilation.

The Bases Changes are provided for information only and will be implemented in accordance with Technical Specification 6.23, "Bases Control Program."

2.12 Summary of Design and Licensing Basis Changes

This Section provides a summary of changes from current design and licensing basis values to revised assumptions for each analysis. The summary is listed in Table 2.12-1 and is broken into segments as detailed in Section 3. The existing analyses for the radiological events 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 RG 1.183 and its supporting documents. In addition, margin was added to various secondary side mass flow parameters in anticipation of possible changes to thermal-hydraulic analyses that define the inputs to the radiological calculations. This accounts for differences in some of the parameters listed in Table 2.12-1.

Table 2.12-1

Summary of Changes to Design/ Licensing Basis For the Radiological Analyses

Parameter	Current Basis	Proposed Basis	Reason for Change
Alternate Source Term (Section 3.1)			
EAB Dose	First 2 hours of accident	Worst 2 hours of accident	RG 1.183
Offsite Breathing Rates			RG 1.183
0 – 8 hours	3.47E-04 (m^3/sec)	3.50E-04 (m^3/sec)	
8 – 24 hours	1.75E-04 (m^3/sec)	1.80E-04 (m^3/sec)	
24 – 720 hours	2.32E-04 (m^3/sec)	2.30E-04 (m^3/sec)	
Dose Conversion Factors	ICRP30	FGR 11 and 12	RG 1.183
RCS and Secondary Side Technical Specification Activity	Current	Table 3.3-1	Revised to reflect increased operational leakage and clean-up. Identified by Westinghouse (Reference 21).
Pre-accident Iodine Spike Activities	Current	Table 3.3-2	
Concurrent Iodine Spike Appearance Rate	Current	Table 3.3-3	
Millstone Unit 2 Control Room (Section 3.1.5.4)			
Unfiltered In-leakage	130 cfm	200 cfm	Proposed TS change. New license/ design basis value
Time to isolate the Control Room (including instrument and damper response time)	10 seconds	20 seconds	Change in analysis assumption – allow operational flexibility
Operator Action: Time after isolation to align ventilation to filtered recirculation mode	10 minutes	1 hour	Proposed license/ design basis change- Analysis assumption to reduce operator burden

Table 2.12-1

Summary of Changes to Design/ Licensing Basis For the Radiological Analyses

Parameter	Current Basis	Proposed Basis	Reason for Change
X/Q's (unchanged for Millstone stack)	Murphy & Campe	ARCON96 (listed in Table 1.3-4)	ARCON96
Breathing Rate	3.47E-04 m^3/sec	3.5E-04 m^3/sec	RG 1.183
Loss-of Coolant Accident (Section 3.1)			
Release Timing	RG 1.4	RG 1.183	RG 1.183
Core Release Fractions	RG 1.4	RG 1.183	RG 1.183
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	RG 1.183
Iodine Chemical Form in Containment Sump & RWST	5% Cesium Iodide 91% Elemental Iodine 4% Organic Iodine	97% Elemental Iodine 3% Organic Iodine	RG 1.183
Spray start time	101 seconds	75 seconds	Updated containment analyses
Mixing Rates, unsprayed to sprayed volume	101 - 303 sec: 6.06 hr^{-1} 303 - 455 sec: 7.76 hr^{-1} 455- 7460 sec: 6.34 hr^{-1}	2 hr^{-1} until sprays secured	RG 1.183
Spray Coverage	75.08%	35.4%	More conservative evaluation method
Spray Removal Coefficient	Elem: 20 hr^{-1} Part: 3.03 hr^{-1}	Elem: 20 hr^{-1} Part: 6.42 hr^{-1}	Updated analyses
Spray Effectiveness Time	Elem: to 0.715 hrs Part: to 1.58 hrs	Elem: to 3.03 hrs Part: to 3.23 hrs	Updated analysis
Containment Bypass Leak Rate	0.0072 La	0.014 La	Proposed TS change to allow operational flexibility
ECCS Leakage Initiation	25 minutes	27.5 minutes	Updated analysis

Table 2.12-1

Summary of Changes to Design/ Licensing Basis For the Radiological Analyses

Parameter	Current Basis	Proposed Basis	Reason for Change
Containment Sump Volume	2.86E+05 gallons	2.822E+05 gallons	Added conservatism – smaller volume yields higher concentration
RWST Release Rate	3.32 cfm	3.5 cfm	Updated analysis
Fuel Handling Accident (Section 3.2)			
See Control Room Changes Listed Earlier in This Table			
Steam Generator Tube Rupture Accident (Section 3.3)			
Steam Generator Mass Releases and Timing	FSAR Table 14.6.3-4	Table 3.3-4	Mass release increased for conservatism. Increased duration for entry to shutdown cooling.
RCS Mass Releases and Timing	FSAR Table 14.6.3-4	Table 3.3-4	Mass release increased for conservatism. Timing change to maximize control room dose.
Initial Steam Generator Mass (for determining inventory)	134,000 lbm per SG	280,000 lbm per SG	Analysis assumption to maximize released activity.
Minimum Steam Generator Mass	100,000 lbm per SG	80,000 lbm per SG	Analysis assumption to reduce holdup time.
RCS Mass	430,000 lbm	423,000 lbm	More accurate value used. Represents a transient minimum.

Table 2.12-1

Summary of Changes to Design/ Licensing Basis For the Radiological Analyses

Parameter	Current Basis	Proposed Basis	Reason for Change
Primary to Secondary Leak Rate	0.035 GPM	150 GPD (0.104 GPM)*	Two times the proposed TS change - Leak rate increased for operational flexibility
Moisture Carryover in Steam Generators	1%	0.4%	Reduction in extent of conservatism (design value is 0.2%)
Iodine Spiking factor for SGTR event	500	335	RG 1.183
Duration of Iodine Spike	4 hours	8 hours	RG 1.183
Duration of Release From Intact Steam Generator	16 hours	17 hours	Intact SG – conservatively added 1 hour to identify and isolate affected SG.
Main Steam Line Break Accident (Section 3.4)			
Steam Generator Contents	100,000 lbm	91,092 lbm (minimum – used as holdup mass in intact SG) 248,891 lbm (maximum - used to maximize inventory)	More accurate value results in smaller retention volume. Analysis assumption to maximize released activity.
RCS Mass	430,000 lbm	428,400 lbm	More accurate value – results in higher specific activity with failed fuel.
Primary to Secondary Leak Rate	0.035 GPM	150 GPD (0.104 GPM)*	Two times the proposed TS change - Leak rate increased for operational flexibility.

Table 2.12-1

Summary of Changes to Design/ Licensing Basis For the Radiological Analyses

Parameter	Current Basis	Proposed Basis	Reason for Change
Duration of Release From Affected Steam Generator	24 hours	36 hours	Affected SG – increased duration to reflect Tech Spec time to cold shutdown.
Peaking Factor	1.65	1.69	Conservative analysis assumption – added margin.
Operator Actions		New Action to isolate the control room within 4 hours	Proposed license/ design basis change. Analysis assumption – necessary to meet control room dose criteria.
Control Rod Ejection Accident (Section 3.5)			
Containment Bypass Leak Rate	0.0072 La	0.014 La	Proposed TS change - Analysis assumption to allow operational flexibility
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	RG 1.183
Iodine Chemical Form released from Steam Generator	5% Cesium Iodide 91% Elemental Iodine 4% Organic Iodine	97% Elemental Iodine 3% Organic Iodine	RG 1.183
Peaking Factor	1.65	1.69	Analysis assumption to add margin
Core Fraction in the Gap	10% Noble Gases 30% Kr-85 10% Iodines 12% I-131	10% Noble Gases 10% Iodines	RG 1.183

Table 2.12-1

Summary of Changes to Design/ Licensing Basis For the Radiological Analyses

Parameter	Current Basis	Proposed Basis	Reason for Change
Primary to Secondary Leak Rate	0.035 GPM (50.4 GPD)	150 GPD (0.104 GPM)*	Two times the proposed TS change - Leak rate increased for operational flexibility

* 150 GPD is used as the design basis input to support a 75 GPD leak rate limit in consideration of Generic Letter 2006-01 and TSTF-449.

Control Room filter efficiency values of ≤ 90% for elemental and aerosol iodines and 70% organic iodine were used in the control room assumptions. These values are lower than the design basis values of 90% for all iodine species but are used only to increase conservatism in the analyses and not to change the current design basis efficiencies. The TEDE values for the MPS2 control room presented in the license amendment request are conservative using the lower efficiency filter values. While the filter shine dose from higher efficiency filters might increase only slightly, the inhalation dose from using lower efficiency filters results in a very conservative value that outweighs any slight increase.

In addition, credited filter efficiencies for the enclosure building filtration system (EBFS) of 70% for all iodine species were used in the analyses. These values are lower than the design basis values of 90% for elemental and aerosol iodines and 70% for organic iodine but are used only to increase conservatism in the analyses and not to change the current design basis efficiencies.

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
- Control Rod Ejection Accident (CREA).

These analyses have incorporated the features of the 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 EAB for the worst 2-hour period, and for the LPZ and MPS2 control room for the duration of the accident (30 days). DNC 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
Concurrent 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
Concurrent Iodine Spike	5 rem TEDE	2.5 rem TEDE
Control Rod Ejection Accident	5 rem TEDE	6.3 rem TEDE
Fuel Handling Accident	5 rem TEDE	6.3 rem TEDE

3.1 Design Basis 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
- ECCS Component Leakage
- Refueling Water Storage Tank Vent
- Shine from the plume
- Shine from containment
- Shine from the control room filter loading

Doses are calculated at the EAB for the worst-case two-hour period, at the LPZ, and in the MPS2 control room. The methodology used to evaluate the control room and offsite doses resulting from a LOCA is 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 is 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 an 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% (immediate 50% plateout) 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 applies 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 ORIGENS code. ORIGENS is part of the SCALE computer code system (Reference 6). Table 3.1-3 lists the 62 isotopes and the associated curies at the end of a fuel cycle that was input to RADTRAD-NAI. The reactor core in MPS2 consists of 217 fuel assemblies with various Uranium-235 enrichments. The current licensed maximum value for fuel enrichment is 5.0% (Technical Specification 5.3.1). Both a 3.5% average enriched core and a 5% average enriched core (representing the maximum enrichment) were evaluated.

Fuel assemblies with three different burnups, representing one (1), two (2) and three (3) cycles, were assumed to determine an equilibrium core source term at the end of a fuel cycle. Different combinations of low and high core burnups (up to a core average burnup of 50,000 MWD/MTU) and enrichments were used to determine a conservative source term that maximized dose due to its predominance, release fraction and dose conversion factor (DCF). Also shown

in Table 3.1-3 are 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	1.194E+06	1.190E-16	0.000E+00
Kr-85m	Noble gas	2.451E+07	7.480E-15	0.000E+00
Kr-87	Noble gas	4.860E+07	4.120E-14	0.000E+00
Kr-88	Noble gas	6.865E+07	1.020E-13	0.000E+00
Xe-133	Noble gas	1.569E+08	1.560E-15	0.000E+00
Xe-135	Noble gas	5.658E+07	1.190E-14	0.000E+00
Xe-138	Noble gas	1.316E+08	5.770E-14	0.000E+00
I-131	Halogen	7.719E+07	1.820E-14	8.890E-09
I-132	Halogen	1.105E+08	1.120E-13	1.030E-10
I-133	Halogen	1.504E+08	2.940E-14	1.580E-09
I-134	Halogen	1.666E+08	1.300E-13	3.550E-11
I-135	Halogen	1.407E+08	7.980E-14	3.320E-10
Rb-86	Alkali Metal	2.693E+05	4.810E-15	1.790E-09
Rb-88	Alkali Metal	6.939E+07	3.360E-14	2.260E-11
Rb-89	Alkali Metal	9.005E+07	1.060E-13	1.160E-11
Cs-134	Alkali Metal	2.821E+07	7.570E-14	1.250E-08
Cs-136	Alkali Metal	7.545E+06	1.060E-13	1.980E-09
Cs-137	Alkali Metal	1.319E+07	7.740E-18	8.630E-09
Cs-138	Alkali Metal	1.437E+08	1.210E-13	2.740E-11
Sb-127	Tellurium	9.663E+06	3.330E-14	1.630E-09
Sb-129	Tellurium	2.694E+07	7.140E-14	1.740E-10
Te-127	Tellurium	9.588E+06	2.420E-16	8.600E-11

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-127m	Tellurium	1.287E+06	1.470E-16	5.810E-09
Te-129	Tellurium	2.653E+07	2.750E-15	2.090E-11
Te-129m	Tellurium	3.943E+06	1.550E-15	6.470E-09
Te-131m	Tellurium	1.161E+07	7.010E-14	1.730E-09
Te-132	Tellurium	1.084E+08	1.030E-14	2.550E-09
Sr-89	Ba -Sr	9.426E+07	7.730E-17	1.120E-08
Sr-90	Ba -Sr	9.627E+06	7.530E-18	3.510E-07
Sr-91	Ba -Sr	1.122E+08	3.450E-14	4.490E-10
Sr-92	Ba -Sr	1.173E+08	6.790E-14	2.180E-10
Ba-139	Ba -Sr	1.388E+08	2.170E-15	4.640E-11
Ba-140	Ba -Sr	1.352E+08	8.580E-15	1.010E-09
Mo-99	Noble Metal	1.427E+08	7.280E-15	1.070E-09
Rh-105	Noble Metal	9.246E+07	3.720E-15	2.580E-10
Ru-103	Noble Metal	1.323E+08	2.250E-14	2.420E-09
Ru-105	Noble Metal	1.011E+08	3.810E-14	1.230E-10
Ru-106	Noble Metal	6.438E+07	0.000E+00	1.290E-07
Tc-99m	Noble Metal	1.250E+08	5.890E-15	8.800E-12
Ce-141	Cerium	1.302E+08	3.430E-15	2.420E-09
Ce-143	Cerium	1.232E+08	1.290E-14	9.160E-10
Ce-144	Cerium	1.046E+08	8.530E-16	1.010E-07
Np-239	Cerium	1.980E+09	7.690E-15	6.780E-10
Pu-238	Cerium	6.162E+05	4.880E-18	7.790E-05
Pu-239	Cerium	3.656E+04	4.240E-18	8.330E-05
Pu-240	Cerium	6.590E+04	4.750E-18	8.330E-05
Pu-241	Cerium	1.496E+07	7.250E-20	1.340E-06

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
Am-241	Lanthanides	1.916E+04	8.180E-16	1.200E-04
Cm-242	Lanthanides	6.566E+06	5.690E-18	4.670E-06
Cm-244	Lanthanides	1.102E+06	4.910E-18	6.700E-05
La-140	Lanthanides	1.359E+08	1.170E-13	1.310E-09
La-141	Lanthanides	1.275E+08	2.390E-15	1.570E-10
La-142	Lanthanides	1.251E+08	1.440E-13	6.840E-11
Nb-95	Lanthanides	1.400E+08	3.740E-14	1.570E-09
Nd-147	Lanthanides	4.995E+07	6.190E-15	1.850E-09
Pr-143	Lanthanides	1.233E+08	2.100E-17	2.190E-09
Y-90	Lanthanides	1.004E+07	1.900E-16	2.280E-09
Y-91	Lanthanides	1.171E+08	2.600E-16	1.320E-08
Y-92	Lanthanides	1.176E+08	1.300E-14	2.110E-10
Y-93	Lanthanides	1.300E+08	4.800E-15	5.820E-10
Zr-95	Lanthanides	1.389E+08	3.600E-14	6.390E-09
Zr-97	Lanthanides	1.273E+08	9.020E-15	1.170E-09

3.1.3 Determination of Atmospheric Dispersion Factors (X/Q)

3.1.3.1 Millstone Unit 2 Control Room X/Q

The onsite atmospheric dispersion factors were calculated by DNC using the ARCON96 code (Reference 5) and guidance from RG 1.194 (Reference 10). Site meteorological data taken over the years 1997-2001 were used in the calculations and have been provided via Reference 11. Additional information to support ARCON96 inputs is included in Enclosure 1 of this Attachment. The control room X/Qs were calculated for the LOCA for the following MPS2 source points:

- Ground level from the enclosure building/ containment
- RWST vent

The control room X/Q's from the Millstone stack are not recalculated using ARCON96. The values are consistent with the current licensing basis approved in Amendment No. 228, dated March 10, 1999 (Reference 13) and are based on RG 1.145 (Reference 19) 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 EAB and 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. Offsite atmospheric dispersion factors were approved in Amendment No. 228, dated March 10, 1999 (Reference 13) to Facility Operating License No. DPR-65 for MPS2.

3.1.4 Determination of Containment Airborne Activity

3.1.4.1 Containment Sprays

The percentage of containment that is covered by spray is 35.4%. The spray system becomes effective at 75 seconds. The mixing rate during spray operation is 2 turnovers of the unsprayed volume per hour.

The elemental and particulate iodine removal coefficients due to sprays are listed in Table 3.1-4. These spray removal coefficients are used until the allowed decontamination factor (DF) is reached (elemental – 200, particulate – 50). Spray removal credit stops for elemental iodine after 3.03 hours and for particulate after 3.23 hours. At that time further iodine removal is ignored due to

sprays even though the spray system may remain operating. A maximum elemental iodine DF of 199 and a maximum particulate iodine DF of 49.5 were calculated during the period that sprays are assumed operating.

3.4.1.2 Natural Deposition

A reduction in airborne radioactivity in the containment by natural deposition within the unsprayed region of 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 is 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 MPS2 to the EAB, LPZ, and MPS2 control room. The QADS code was used to calculate the shine dose to the control room from containment shine and control room filter shine.

3.1.5.2 Basic Data & Assumptions for LOCA (see Table 3.1-4)

Table 3.1-4

Basic Data and Assumptions for LOCA

Parameter or Assumption / (Reference)	Value
Containment Leak Rate: (Technical Specifications)	0.5% by weight of the containment air per 24 hours (L_a)
Containment Bypass Leak Rate:	$0.014L_a$
Containment Leak Rate Reduction:	50% after 24 hours
Secondary Containment Drawdown Time:	110 seconds
Iodine Chemical Form in Containment Atmosphere: (Reference 1)	95% Cesium Iodide 4.85% Elemental Iodine 0.15% Organic Iodine

Table 3.1-4

Basic Data and Assumptions for LOCA

Parameter or Assumption / (Reference)	Value
Iodine Chemical Form in the Sump and RWST: (Reference 1)	97% Elemental 3% Organic
Containment Sump pH:	≥ 7
Dose Conversion Factors:	References 8 and 9
EBFS Filter Efficiency:	70% of all Iodines and Particulates*
Spray System Effective Period of Operation:	75 sec – 3.03 hrs (elemental) 75 sec – 3.23 hrs (particulate)
Elemental Iodine Removal Coefficient:	20 per hour
Particulate Iodine Removal Coefficient	DF < 50: 6.42 DF $\geq$ 50: not credited
Mixing Rate	2 per hour
Free Air Volume of Containment:	1,899,000 ft^3
Sprayed Volume Percentage	35.4%
ECCS System Leakage Outside Containment(2 x program limits):	24 gallons per hour
ECCS Leakage Start Time	27.5 minutes
Percentage of Iodines Released from ECCS water	10%
Containment Sump Minimum Volume	2.822E+05 gallons (= 3.773E+04 ft^3)
RWST Minimum Volume:	6,469 gallons
RWST Maximum Volume:	475,000 gallons

* 70% efficiency is conservatively used in the analysis for all iodine species. Actual design basis is 90% elemental/aerosol and 70% organic.

3.1.5.3 Containment Leakage Model

The containment leakage consists of filtered and bypass leakage. The bypass leakage bypasses the secondary containment and is released unfiltered at ground level directly from containment. As stated in the data and assumptions, the total containment leak rate (La) is 0.5% per day. The entire containment leak rate bypasses the secondary containment until EBFS produces a negative pressure of >0.25" W.G. in the secondary containment at 110 seconds at which point the leak rate is 0.007% per day (0.014La). The leak rate is reduced by one-half (0.0035% per day) at 24 hours. The filtered leak rate is (La – Bypass) or 0.493% per day after 110 seconds and half that after 24 hours (0.2465%/day).

3.1.5.4 Model of ECCS Leakage

The ECCS fluid consists of the contaminated water in the sump of the containment. With the exception of noble gases, this water contains all the fission products released from the core. All radioactive materials, except iodine, remain in the recirculating sump fluid. Therefore the ECCS leakage model consists of 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 spray headers and sprayed back into the containment sump following RWST drawdown. Also, following a design basis LOCA, valve realignment occurs to switch the suction water source for core cooling from the RWST to the containment sump.

ECCS leakage is assumed when ESF systems circulate sump water outside containment. Typical leakage sources are through packing glands, pump shaft seals and flanged connections. The leakage of recirculating sump fluids commences at 27.5 minutes, which is the earliest time of recirculation and is assumed to leak at 24 gallons per hour. This will support a Technical Specification 6.13 (Systems Integrity) program limit for ECCS leakage of 12 gallons per hour.

The temperature of the containment sump does not exceed 212 degrees F. At this maximum temperature, per the guidance of RG1.183, a conservative flash fraction of 0.1 is used for the ECCS leakage during the entire event. The water in the sump is assumed to remain at the minimum volume (3.773E+04 ft3) for the duration of the event to maximize concentration of nuclides.

3.1.5.5 Model of ECCS Back Leakage to the RWST

The RWST back leakage flow paths considered are identified in Table 3.1-5. 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.

Back leakage from the containment sump via the normal ECCS and containment spray pump suction line to the RWST is negligible. The established leakage criterion for the check valves in this suction line prevent contaminated water from reaching the RWST.

The RADTRAD-NAI source term used to model the ECCS leakage into the RWST contains only the iodine isotopes. The RWST backleakage model consists of 40% of the core inventory of iodine (5% released to the sump water during the 30 minute gap release phase and 35% released to the sump water during the following 1.3 hour early in-vessel phase) as being transported from the core to the containment sump. The iodine form is 97% elemental and 3% organic in accordance with RG 1.183.

Using the methodology approved in Reference 22, 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. Timing and flow rates discharged to the RWST from the RWST backleakage paths have been calculated and summarized in Table 3.1-5. 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 3.1-5
Contaminated Inflow to RWST

Source	Time to Reach RWST, hours	Flow Rate, GPM
Leakage through valve CS-51 (Cont. Spray Header "A" test line)	12.89	0.05
Leakage through valve CS-50 (Cont. Spray Header "B" test line)	14.06	0.05
Leakage through valve SI-659/660 (HPSI/ LPSI/ Cont. Spray pump minimum recirculation flow line)	15.27	0.40
Leakage through valve SI-460 (LPSI/ Cont. Spray test line)	16.14	0.20

Table 3.1-6

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

Time (hrs)	Flow Rate (GPM)	Volume (gal)
6.45	0.05	0.00
7.03	0.1	1.75
7.63	0.5	5.38
8.07	0.7	18.5
720.0	0.7	29,919.50

Since the containment sump water will not exceed 212ºF, it is assumed that 10% of the iodine in the sump water that leaks back to the RWST will be released to the environment at the RWST airflow rate per the guidance of RG 1.183. The RWST airflow rate of 3.5 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 6.45 hours post-LOCA.

3.1.5.6 Millstone Unit 2 Control Room

The MPS2 control room volume is 35,656 ft^3. The LOCA causes a safety injection actuation signal (SIAS) to isolate the control room. The control room is isolated within 20 seconds after a SIAS. 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.

An operator action to ensure proper alignment and the operation of the control room emergency ventilation system for the filtered recirculation mode is credited 1 hour after isolation.

The post LOCA dose consequences from external sources to the MPS2 control room are due to the following 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.

Table 3.1-7 lists TEDE to the EAB and LPZ from a LOCA at MPS2. The dose to the EAB and LPZ is less than the 25 rem TEDE limit stated in 10CFR50.67 and RG 1.183. The EAB dose represents the worst 2-hour dose for each release pathway.

3.1.6 LOCA Analysis Results

The dose to the MPS2 control room is less than the 5 rem TEDE limit specified in 10 CFR 50.67 and RG 1.183.

Table 3.1-7

TEDE from a Millstone Unit 2 LOCA

Location	TEDE (rem)
EAB	2.9E+00
LPZ	1.7E+00
Millstone Unit 2 Control Room	3.0E+00

3.2 Fuel Handling Accident (FHA)

This section describes the methods employed and results of the FHA design basis radiological analysis. The analysis includes doses associated with release of gap activity from a fuel assembly either inside containment or from the spent fuel pool. Doses were calculated at the EAB, at the LPZ boundary, and in the MPS2 control room. The methodology used to evaluate the control room and offsite doses resulting from the FHA: 1) is consistent with RG 1.183 in conjunction with TEDE radiological units and limits; 2) uses ARCON96 based onsite atmospheric dispersion factors; and 3) uses Federal Guidance Reports No. 11 and 12 dose conversion factors.

The FHA was previously approved by the NRC via Reference 20 for selective implementation of the AST. The analysis in this amendment request differs from the previous analysis by the following:

1) Revised Control Room X/Qs (now based on ARCON96).
2) Changes were made to control room in-leakage assumptions (from 130 scfm to 200 scfm).
3) Reduced control room filtration efficiency (from 90% to 70% for organic only).
4) Increased time for automatic isolation of control room, from 10 seconds to 20 seconds, based on control room ventilation radiation monitor response.
5) Increased time for operator action following control room isolation, from 10 minutes to 1 hour, to place the control room emergency ventilation in filtered recirculation mode.
6) The cask drop accident will be referred to as the cask tip accident.

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. 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 spent fuel pool (located in the auxiliary 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 auxiliary building.

The cask drop accident discussed in the UFSAR that has radiological consequences will be referred to as a cask tip accident to avoid confusion. This accident also bounds the consequences associated with drop of a consolidated fuel canister which can contain up to 2 consolidated fuel assemblies with a minimum of 5 years decay. This scenario has the cask tipping over in the spent

fuel pool and damaging impacted fuel assemblies. It is postulated that 1560 fuel assemblies are damaged. Of these, 184 have a decay time of 1 year and the remainder for 5 years. Administrative controls limit the age of fuel assemblies in the area of potential impact.

3.2.2 FHA Source Term Definition

In accordance with Regulatory Position 3 of RG 1.183 the core source was determined using ORIGENS to evaluate multiple cycle designs (based on the DNC fuel management scheme for enrichment and burnup). The core inventory used was approved in the FHA selective implementation of the AST in Amendment 284 (Reference 20) and is described in the LOCA scenario (Section 3.1.2). Gap fractions used in the FHA and cask tip analyses differ from Regulatory Position 3 due to a small number of fuel rods having the potential to exceed the linear heat generation rate criteria in Footnote 11 of RG 1.183. As a result, new gap fractions were proposed in Reference 23 and approved in Amendment 284. They are listed in Table 3.2-1.

For the FHA analyses, the core inventory was used to calculate the gap activity of one fuel assembly 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 auxiliary building is the release point.

For the cask tip accident, the core inventory was used to determine the fuel assembly inventory. The fuel assembly inventory was decayed to get the 1 or 5 year decay which was applied to 184 and 1376 assemblies, respectively. The gap activity of these assemblies was released. This release occurs from the auxiliary building.

3.2.3 FHA Release Transport

This evaluation does not credit operability or operation of the containment purge system, or auxiliary building ventilation. This evaluation assumes that the personnel hatch, equipment hatch, auxiliary building roll-up door and penetrations are open for the duration of the 2 hour release. Therefore, any release out of the auxiliary building or containment is a ground level release.

Releases from the auxiliary building or containment to the environment are at a rate of 3.5 air changes per hour. This assures that essentially all of the activity in the auxiliary building and containment analyses are 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 auxiliary building or containment. Additionally, no

credit is taken for dilution or mixing of the activity released to the fuel building or containment air volumes.

The release from a FHA inside the containment is assumed unfiltered and is released via a ground level release from the enclosure building.

For a FHA, or cask tip accident in the auxiliary building, the release is also assumed unfiltered via a ground level release from the enclosure building. The offsite and control room X/Qs are the same for a FHA in containment or the auxiliary building, both conservatively assume the release is from the enclosure building.

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 DNC 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 (provided in Reference 11) were used in the evaluations. Control room X/Q values were calculated at the enclosure building edge nearest the control room. Parameters and assumptions utilized to perform the ARCON96 analysis are shown in Enclosure 1 to Attachment 1. The control room atmospheric dispersion factors are 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 have not changed and are reported in Table 1.3-3.

3.2.5 FHA Analysis Assumptions & Key Parameter Values

The basic data and assumptions for the FHA are listed below in Table 3.2-1. The basic data and assumptions for the cask tip accident are listed below in Table 3.2-2.

Table 3.2-1
Data and Assumptions for the Fuel Handling Accident Analysis

Data / Assumption	**Value**
Gap Fractions	I-131: 12% Kr-85: 30% Other Noble Gases: 10% Other Halogens: 10%
Pool Decontamination Factor:	Noble Gases: 1 Iodines: 200 (effective DF)
Release Point:	Enclosure Building/ Containment Ground
Decay Time:	100 hours
Radial Peaking Factor:	1.83
Duration of Release to the Environment:	2 hours
Fuel Damage:	1 assembly
Control Room Isolation Time	20 seconds after start of release

Table 3.2-2
Data and Assumptions for the Cask Tip Accident Analysis

Data / Assumption	**Value**
Gap Fractions	I-131: 12% Kr-85: 30% Other Noble Gases: 10% Other Halogens: 10%
Pool Decontamination Factor:	Noble Gases: 1 Iodines: 200 (effective DF)
Release Point:	Enclosure Building/ Containment Ground
Fuel Damage:	1560 assemblies total
Decay Time:	1 year – 184 assemblies 5 years – 1376 assemblies
Radial Peaking Factor:	1

Table 3.2-2
Data and Assumptions for the Cask Tip Accident Analysis

Data / Assumption	Value
Duration of Release to the Environment:	2 hours
Control Room Isolation Time	20 seconds after start of release
Filter and plume shine are negligible considering the isotopes involved (I-129 and Kr-85)	NA

3.2.6 FHA Analysis Results

The offsite and control room dose summary due to submersion, inhalation and shine is listed below. The MPS2 FHA assumes a two-hour release without building integrity or filtered release pathway for the containment and spent fuel pool FHA. The associated worst case TEDE is presented in Table 3.2-3. The associated worst case TEDE for the cask tip accident is presented in Table 3.2-4. All doses are less than those limits specified in RG 1.183 and 10 CFR 50.67.

Table 3.2-3

Dose Summary for the Fuel Handling Accident Analysis

Location	TEDE (rem)	Limits (rem)
EAB	1.5E+00	6.3
LPZ	2.0E-01	6.3
Millstone Unit 2 Control Room	3.1E+00	5

Table 3.2-4

Dose Summary for the Cask Tip Accident Analysis

Location	TEDE (rem)	Limits (rem)
EAB	5.0E-01	6.3
LPZ	5.0E-02	6.3
Millstone Unit 2 Control Room	2.5E-01	5

3.3 Steam Generator Tube Rupture Accident

This section describes the methods employed and the results of the 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 EAB, at the LPZ, and in the control room. The methodology used to evaluate the control room and offsite doses resulting from the SGTR accident is consistent with RG 1.183 in conjunction with TEDE radiological units and limits, uses ARCON96 based onsite atmospheric dispersion factors, and 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) or main steam safety valves (MSSVs). The ADV/ MSSVs on the affected steam generator are assumed to open to control steam generator pressure at the beginning of the event. No credit is taken for release through the condenser due to loss of offsite power. The affected generator discharges steam to the environment for 1 hour until the generator is isolated. Break flow into the affected steam generator continues until the RCS is at a lower pressure. Additional release from the affected steam generator is modeled following isolation to complete depressurization of the steam generator early in the event to maximize the dose consequences. Depressurization of the steam generator may be necessary to initiate shutdown cooling.

The intact generator discharges steam for a period of approximately 17 hours until the primary system has cooled sufficiently to allow a switchover to the shutdown cooling (SDC) system. At this point, SDC can remove all the decay heat to achieve cold shutdown and steaming is no longer required for cooldown. No fuel damage is predicted as a result of a SGTR. Therefore, the SGTR analysis was evaluated for a pre-accident iodine spike and a concurrent accident iodine spike.

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 level of activity. The thermo-hydraulic analysis of the SGTR accident indicates that no additional fuel 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.

RG 1.183 indicates 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 required to be used in the analysis by RG 1.183. Primary side concentration is based on the technical specification limits of 100/ E_{bar} for gross gamma and 1.0 uCi/gm DEQ I-131 for iodines. Secondary side concentration is based on the technical specification limit of 0.1 uCi/gm DEQ I-131 for iodine. Since there is not a technical specification limit for gross gamma activity, a secondary side gross gamma concentration was developed to ensure that a suitably conservative source term was used.

RG 1.183 also dictates 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 MPS2, the maximum iodine concentration allowed by technical specifications as the result of an iodine spike is 60 uCi/gm DEQ I-131. This value is treated as the pre-accident iodine spike and is listed in Table 3.3-2. RG 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 to account for increased reactor coolant leakage and clean-up (Reference 21).

The dose conversion factors used to calculate the TEDE doses and DEQ I-131 for the SGTR accident were taken from Table 3.1-3 for the isotopes required by RG 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
I-131	5.62E-01	5.92E-02
I-132	3.10E+00	1.90E-01
I-133	1.85E+00	1.83E-01
I-134	5.14E+00	1.83E-01
I-135	3.66E+00	3.11E-01
Kr-85m	4.90E+00	0.00E+00
Kr-85	1.10E+01	0.00E+00
Kr-87	4.60E+00	0.00E+00
Kr-88	8.58E+00	0.00E+00
Xe-131m	2.12E+01	0.00E+00
Xe-133m	2.12E+00	0.00E+00
Xe-133	7.75E+01	0.00E+00
Xe-135m	3.99E+00	0.00E+00
Xe-135	2.60E+01	0.00E+00
Xe-138	3.68E+00	0.00E+00
Br-84	4.85E-01	3.32E-06
Co58	1.11E-01	6.43E-06
Co-60	1.27E-02	7.43E-07
Rb-88	5.72E+00	2.75E-05
Sr-89	3.37E-03	1.93E-07
Sr-90	2.88E-04	1.66E-08
Sr-91	2.60E-02	1.13E-06
Y-91m	1.38E-02	1.58E-07

Nuclides	RCS, uCi/gm	Secondary Side Water, uCi/gm
Y91	1.25E-04	7.11E-09
Y93	1.13E-01	4.80E-06
Zr95	9.38E-03	5.42E-07
Nb95	6.74E-03	3.73E-07
Mo99	1.58E-01	8.79E-06
Tc99m	1.31E-01	4.63E-06
Ru103	1.81E-01	1.05E-05
Ru106	2.16E+00	1.25E-04
Te129m	4.58E-03	2.64E-07
Te129	7.16E-01	1.07E-05
Te131m	3.82E-02	1.97E-06
Te131	2.34E-01	1.47E-06
Te132	4.19E-02	2.31E-06
Cs134	9.98E-02	1.04E-05
Cs136	1.26E-02	1.29E-06
Cs137	1.32E-01	1.38E-05
Ba140	3.14E-01	1.77E-05
La140	6.29E-01	3.34E-05
Ce141	3.61E-03	2.07E-07
Ce143	7.10E-02	3.63E-06
Ce144	9.61E-02	5.41E-06
Np239	5.47E-02	2.97E-06

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
I-131	5.62E-01	3.37E+01
I-132	3.10E+00	1.86E+02
I-133	1.85E+00	1.11E+02
I-134	5.14E+00	3.09E+02
I-135	3.66E+00	2.19E+02

Table 3.3-3

Concurrent Iodine Spike Appearance Rate

Nuclide	Appearance rate for 1 uCi/gm DEQ I-131, uCi/sec	Spike = 335, SGTR Appearance Rate, uCi/sec
I-131	2.85E+03	9.56E+05
I-132	6.95E+04	2.33E+07
I-133	1.26E+04	4.23E+06
I-134	2.62E+05	8.78E+07
I-135	4.02E+04	1.35E+07

3.3.3 Release Transport

3.3.3.1 Affected Steam Generator

The source term resulting from the radionuclides in the primary system coolant and from the iodine spiking in the primary system is transported through the affected steam generator by the break flow. 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 a result of flashing pass directly to the environment through the ADVs. The remainder of the break flow enters the steam generator liquid. Releases of radionuclides in the steam generator liquid and those entering the steam generator from the

unflashed break flow are released as a result of secondary liquid boiling. Moisture carryover results in an allowance for a partition factor of 100 for iodines and 250 for all other non-noble gas isotopes. Thus 1% of the elemental and organic iodines and 0.4% of the particulates are released from the steam generator liquid to the environment along with the steam flow. All noble gases are released from the primary system to the environment without reduction or mitigation. The transport model utilized for iodine and particulates is consistent with Appendix E of RG 1.183.

FSAR Section 14.6.3.5 discusses the operator actions supporting this event.

3.3.3.2 Intact Steam Generator

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 generator by the assumed design basis leak-rate of 150 GPD (0.104 GPM) versus that specified in the proposed technical specification change (75 GPD). The design basis leak rate of 150 GPD is used in consideration of Generic Letter 2006-01 to provide a conservative margin between operational leakage limits and design basis. 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 iodines and 250 for all other non-noble gas isotopes. Thus 1% of the iodines and 0.4% of the particulates are assumed to pass into the steam space and then 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 17 hours, after which the SDC system can remove 100% of decay heat with no requirement for steaming to augment cooldown.

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 using the ARCON96 code and guidance from RG 1.194 (Reference 10) for the control room. Site meteorological data taken during the years 1997-2001 (provided in Reference #11) were used in the evaluations. Parameters and assumptions utilized to perform the ARCON96 analysis are shown in Enclosure 1 of Attachment 1. The control room atmospheric dispersion factors are presented in Table 1.3-4.

3.3.4.2 Offsite Atmospheric Dispersion Factors

The EAB and LPZ values used in the SGTR analysis are unchanged and listed in Table 1.3-3.

3.3.5 SGTR Key Analysis Assumptions and Inputs

The basic data and assumptions are listed below in Table 3.3-4. 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	150 GPD (intact steam generator)
Release from secondary side is coincident with loss of off-site power	
Release points: (On loss of off-site power, the condenser is not available for cooling, an ADV on the affected steam generator is assumed as the release pathway instead of MSSVs because of higher X/Qs. Additional cooling is by ADVs on intact steam generator.)	Steam Generator Atmospheric Dump Valves (ADVs)
Assumed Operator Actions: Operator actions are discussed in FSAR Section 14.6.3.5.	
Iodine chemical form (%) released from steam generators to environment	Elemental: 97 Organic: 3
Iodine Partitioning	PC for iodine = 100 (intact SG and non-flashed portion of affected SG leakage) PC for iodine = 1 (flashed portion of affected SG leakage - independent of water level in steam generator)
Moisture Carryover in Intact Steam Generators	0.4% (equivalent to PC = 250 for all other non-noble gas isotopes)

Table 3.3-4

Basic Data and Assumptions for the SGTR Accident

Data / Assumption				
Data / Assumption			**Value**	
RCS Break Flow to Affected Steam Generator				
Time Period		Total Break Flow	Flashed Break Flow	Liquid Break Flow
From	To			
(hour)		lbm	lbm	lbm
0	1	150,000	5,000	145,000
After 1 Hour		51,600	1,200	50,400
Mass Releases to Environment: Affected steam generator			0 to 1 hour (t_{hot} <515°F): 1.700E+05 lbm After 1 hour (SG depressurized): 9.200E+04 lbm	

Mass Releases to Environment: Intact steam generator		
Time, hr		Total ISG ADV/ MSSV Flow, lbm/min
From	To	
0.00	1.00	2.000E+03
1.00	1.11	7.330E+03
1.11	1.71	5.147E+03
1.71	2.33	4.200E+03
2.33	2.74	3.840E+03
2.74	3.18	3.810E+03
3.18	3.72	3.780E+03
3.72	6.50	2.743E+03
6.50	17.61	2.151E+03

Table 3.3-4

Basic Data and Assumptions for the SGTR Accident

Data / Assumption	Value
Maximum ADV Flow Rate:	1.100E+06 lbm/ hr
Steam Generator Liquid Mass	Maximum Initial Mass – 280,000 lbm (used to maximize curies in the SGs) Minimum Transient Mass – 80,000 lbm (used to minimize holdup in steam generators)
Control Room Ventilation Timing (operator action credited to initiate filtered recirculation 1 hour after isolation)	<u>Pre-accident Spike</u> T= 0: start of SGTR T= 20 sec: control room isolation on radiation monitor alarm signal T=1 hr, 20 sec: control room on filtered recirculation <u>Concurrent Spike</u> T= 0: start of SGTR T= 10 min, 20 sec: control room isolation on radiation monitor alarm T= 1 hr, 10 min, 20 sec: control room on filtered recirculation

3.3.6 SGTR Analysis Results

The results of the analyses are presented below for the pre-accident spike (Table 3.3-5) and for the concurrent iodine spike (Table 3.3-6).

Table 3.3-5
Dose Summary for the SGTR Pre-accident Iodine Spike

Location	TEDE (rem)	Limits (rem)
EAB	1.4E+00	25
LPZ	1.8E-01	25
Millstone Unit 2 Control Room	4.5E+00	5

Table 3.3-6
Dose Summary for the SGTR Concurrent Iodine Spike

Location	TEDE (rem)	Limits (rem)
EAB	1.2E+00	2.5
LPZ	1.6E-01	2.5
Millstone Unit 2 Control Room	4.5E+00	5

3.4 Main Steam Line Break Analysis

This section describes the methods employed and results of the MSLB design basis radiological analysis. For the MSLB outside containment no fuel failures are expected and released activity is based on RCS activity at technical specification levels plus iodine spiking (concurrent and pre-accident are evaluated). For MSLB in containment, 3.7% fuel failure is assumed. Doses were calculated at the EAB, at the LPZ, and in the MPS2 control room. The methodology used to evaluate the control room and offsite doses resulting from the MSLB accident is consistent with RG 1.183 in conjunction with TEDE radiological units and limits, uses ARCON96 based onsite atmospheric dispersion factors and Federal Guidance Report No. 11 and 12 dose conversion factors.

3.4.1 MSLB Scenario Description

The MSLB accident begins with a break in one of the main steam lines leading from a steam generator (affected generator) to the turbine coincident with a loss of offsite power. As a result, the condenser is unavailable and cool down of the primary system is through the release of steam to the environment from the intact generator. In order to maximize doses, break scenarios are assessed in the following structures: 1) turbine building, 2) containment and 3) enclosure building.

The intact steam generator has primary to secondary leakage of 150 GPD and steams for approximately 16 hours after which decay heat is removed by the SDC system. The design basis leak rate of 150 GPD is used in consideration of Generic Letter 2006-01 relative to the proposed technical specification leak rate of 75 GPD. The affected steam generator is assumed to go dry immediately and the primary to secondary leak flow is assumed to flash completely and be released to the environment without mitigation for 36 hours, at which time the RCS has cooled down to 212°F and release via this pathway terminates.

3.4.2 MSLB Source Term Definition

The analysis for the MSLB in the turbine building and enclosure building assumes no fuel failure. It uses the primary and secondary liquid source term discussed in Table 3.3-1. It also addresses the pre-accident iodine spike source term discussed in Table 3.3-2 and a concurrent iodine spike at 500 times the appearance rate corresponding to the technical specification limit for normal operation for a period of 8 hours and that source term is listed in Table 3.4-1.

Table 3.4-1

Concurrent Iodine Spike Appearance Rate

Nuclide	Appearance rate for 1 uCi/gm DEQ I-131, uCi/sec	Spike = 500, SGTR Appearance Rate, uCi/sec
I-131	2.85E+03	1.43E+06
I-132	6.95E+04	3.48E+07
I-133	1.26E+04	6.31E+06
I-134	2.62E+05	1.31E+08
I-135	4.02E+04	2.01E+07

The analysis for the MSLB in containment assumes 3.7% fuel failure. This source term is used in conjunction with 1.69 peaking factor and gap fractions as determined by Table 3 of Reg. Guide 1.183. The core inventory is listed in Table 3.1-3.

3.4.3 Release Transport

The source term resulting from radionuclides in the primary system coolant and from iodine spiking in the primary system is transported to the steam generators by an assumed leak rate of 150 gallons per day (0.104 GPM) per steam generator. The maximum amount of primary to secondary leakage currently allowed by the technical specifications to any one steam generator is 0.035 gallons per minute. This leakage of 150 gallons per day (0.104 GPM) per steam generator was assigned to both the affected and intact steam generator.

3.4.3.1 Affected Steam Generator

Three scenarios are considered for break release pathways as discussed in Section 3.4.1 (enclosure building, turbine building and containment). In all 3 cases, the release of secondary side liquid is based on the maximum steam generator volume to maximize isotopic inventory.

For the enclosure building and turbine building scenarios, releases are assumed to pass directly from the building to the environment with no credit taken for holdup, partitioning or scrubbing by the steam generator liquid. The affected steam generator is assumed to steam dry within 750 seconds as a result of the MSLB, releasing all of the nuclides in the secondary coolant that were initially contained in the steam generator. Coincident with the release of secondary side liquid, a 150 GPD primary to secondary leakage occurs with the assumption that

100% of the flow flashes and is released to the environment without mitigation. After 36 hours the RCS will have cooled to below 212°F and the release via this pathway terminates. The transport model utilized for noble gases, iodines and particulates was consistent with Appendix E of RG 1.183.

For the containment scenario, primary to secondary liquid from the affected steam generator is assumed to leak to containment at 150 GPD. After 36 hours the RCS will have cooled to below 212°F and the release to containment terminates. Initially all of the releases from containment bypass the secondary containment at the technical specification limit of 0.5%/day. The enclosure building filtration system becomes effective within 250 seconds (which includes 140 seconds for generation of the actuation signal based on containment high pressure and 110 seconds for drawdown of the enclosure building), after which the release is split into a filtered and bypass portion. The bypass portion is released at ground level unfiltered at the proposed technical specification limit of 0.014 La (0.007%/day). The filtered portion is an elevated release at a rate that corresponds to La - Bypass (0.493%/day). After 24 hours the leak rates are reduced by 50%, consistent with RG 1.183.

3.4.3.2 Intact Steam Generator

The intact steam generator is credited with leaking primary to secondary liquid at 150 gallons per day. This activity is released to the environment via the MSSVs or ADV. The ADV provides the higher X/Q to the control room so it provides the bounding pathway.

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. 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. The minimum steam generator mass is used to minimize holdup and retention of activity. Elemental and organic iodines are released as a function of the partition coefficient of 100 and the steaming rate. Due to moisture carryover, 0.4% 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, noble gas daughters occurs without any mitigation or holdup.

3.4.4 Determination of Atmospheric Dispersion Factors

3.4.4.1 Control Room Atmospheric Dispersion Factors (X/Q)

The onsite atmospheric dispersion factors were calculated by using the ARCON96 code and guidance from RG 1.194 (Reference 10) for the control room. Site meteorological data taken during the years 1997-2001 (provided in Reference #11) were used in the evaluations. Parameters and assumptions utilized to perform the ARCON96 analysis are shown in Enclosure 1 of Attachment 1. The MPS2 control room X/Q values are given in Table 1.3-4. Release points representing sources from a MSLB are: 1) the turbine building blowout panel closest to the control room; 2) the ADV closest to the control room; 3) the enclosure building blowout panel closest to the control room; 4) the MSSVs; and 5) the Millstone stack. The X/Qs for the ADV were used to represent the enclosure building blowout panel closest to the control room. This is conservative and appropriate when considering the ADV is closer to the control room than the blowout panel and it is a point source versus a large area source for the blowout panel.

3.4.4.2 Offsite Atmospheric Dispersion Factors (X/Q)

Offsite X/Qs for ground level releases from the enclosure building are not changed and are listed in Table 1.3-3. They are also applied to the turbine building releases because of near proximity to each other.

Offsite X/Qs for elevated releases from the Millstone stack are also unchanged and are calculated assuming fumigation conditions for 0-2 hour EAB and 0-4 hour LPZ and the highest annual X/Q from all sectors. X/Qs for elevated releases are listed in Table 1.3-3.

3.4.5 MSLB Key Analysis Assumptions and Inputs

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.

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/ Enclosure Bldg/ Containment ADVs / MSSVs
Iodine Partition Coefficients (PC) in Intact Steam Generators:	100
Moisture Carryover in Intact Steam Generators:	0.4%
Primary to secondary Leakage:	Affected SG: 150 GPD Intact: 150 GPD
Steam Generator Liquid Mass:	91,092 lbm (min. volume) 248,891 lbm (max. volume)
Control Room Ventilation Timing:	<u>Enclosure & Turbine Bldg Releases</u> T= 0: start of MSLB T= 4 hr: control room isolation on operator action T= 5 hr: control room on filtered recirculation <u>Containment Release</u> T= 0: start of MSLB T= 140 sec: control room isolation on SIAS (containment high pressure) T=1 hr, 140 sec: control room on filtered recirculation

Table 3.4-2

Basic data and Assumptions for the MSLB Accident

Data / Assumption	Value
Steam Release from Affected Steam Generator: 0 – 750 seconds: 3.08E+05 lbm Primary to secondary Leak for Turbine Bldg and Containment Releases 0 – 36 hours: 0.868 lbm/min (= 150 GPD) Primary to secondary Leak for Enclosure Bldg Releases 0 – 720 hours: 0.868 lbm/min (= 150 GPD)	
Integrated Steam Release from Intact Steam Generator: @ 20 sec: 54,334 lbm* @ 800.1 sec: 54,334 lbm @ 10,000 sec: 534,459.42 lbm @ 60,000 sec: 1,979,171.92 lbm Primary to secondary Leak 0 – 60,000 sec: 0.868 lbm/min (150 GPD)	

*Data for the first 20 seconds represents flow from the intact SG out the break. After 20 seconds, the main steam isolation valve (MSIV) on the intact SG is closed, isolating that portion of the break flow. At 800.1 seconds, releases from the intact SG initiate to allow cooldown and to maximize dose consequences.

3.4.6 MSLB Analysis Results

TEDE to the EAB, LPZ and MPS2 control room from a MPS2 MSLB is summarized below for the concurrent (Table 3.4-3 and 3.4-4 for the enclosure building and turbine building, respectively) and pre-accident spikes (Table 3.4-5 and 3.4-6 for the enclosure building and turbine building, respectively) and for the fuel failure in the containment case (Table 3.4-7). The concurrent spike results in the highest dose consequences for both offsite and the control room. All doses are within the limits specified in RG 1.183 and 10 CFR 50.67.

Table 3.4-3

TEDE from MSLB: Concurrent Iodine Spike – Enclosure Building

Location	TEDE (rem)	Limits (rem)
EAB	1.6E-01	2.5
LPZ	5.4E-02	2.5
Millstone Unit 2 Control Room	3.8E+00	5

Table 3.4-4

TEDE from MSLB: Concurrent Iodine Spike – Turbine Building

Location	TEDE (rem)	Limits (rem)
EAB	1.6E-01	2.5
LPZ	5.4E-02	2.5
Millstone Unit 2 Control Room	4.7E+00	5

Table 3.4-5

TEDE from MSLB: Pre-accident Iodine Spike – Enclosure Building

Location	TEDE (rem)	Limits (rem)
EAB	9.1E-02	25
LPZ	2.8E-02	25
Millstone Unit 2 Control Room	2.6E+00	5

Table 3.4-6

TEDE from MSLB: Pre-accident Iodine Spike – Turbine Building

Location	TEDE (rem)	Limits (rem)
EAB	9.1E-02	25
LPZ	2.9E-02	25
Millstone Unit 2 Control Room	4.0E+00	5

Table 3.4-7

TEDE from MSLB: 3.7% Fuel Failure – Containment

Location	TEDE (rem)	Limits (rem)
EAB	1.4E-01	25
LPZ	4.2E-02	25
Millstone Unit 2 Control Room	2.0E+00	5

3.5 Control Rod Ejection Accident (CREA) Analysis

This section describes the evaluation of TEDE at the EAB, LPZ and MPS2 control room from a MPS2 CREA 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. Each release path is considered independently as the only one available. The actual doses for the accident would be a composite of doses resulting from portions of the release going out the two different pathways. When regulatory compliance to dose limits can be demonstrated for each of the cases, the consequences of a case that is a combination of the two will be encompassed by the more restrictive of the two analyzed cases.

3.5.1 CREA Scenario Description

This accident is defined as the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of a control rod. 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.

3.5.2 CREA Source Term Definition

The core source term used in the CREA analysis is 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, the fractions of core inventory assumed to be in the pellet-to-clad gap and the peaking factor.

The analysis is based on the assumption that 11.5% failed fuel occurs during a CREA.

3.5.3 Release Transport

Two release paths are considered for the CREA: 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:

- The core release fractions are based on Appendix H of RG 1.183.
- Containment sprays are not credited for iodine removal. ECCS leakage and RWST back leakage are not considered.
- Natural deposition in the containment is not assumed.

Time to isolate the control room following a CREA is 105 seconds. This is based on the safety injection signal initiated 1 minute after a CREA, followed by a 25 second delay due to diesel startup for the enclosure building filtration actuation signal to initiate isolation and then 20 seconds for the control room to isolate.

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 assumed primary to secondary leak rate of 150 GPD exists until shutdown cooling is in operation and releases from the steam generators terminate at approximately 16 hours. 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 steam flow is released via the MSSVs/ADVs during the CREA until entry into the shutdown cooling window. Steam flows are provided in Table 3.5-1.

3.5.4 Determination of Atmospheric Dispersion Factors (X/Q)

3.5.4.1 Control Room Atmospheric Dispersion Factors (X/Q)

The onsite atmospheric dispersion factors were calculated by using the ARCON96 code and guidance from RG 1.194 (Reference 10) for the control room. Site meteorological data taken during the years 1997-2001 (provided in Reference 11) were used in the evaluations. Parameters and assumptions utilized to perform the ARCON96 analysis are shown in Enclosure 1 of Attachment 1. Release points representing sources from a CREA are: 1) the enclosure building; 2) the site stack; 3) the ADV closest to the control room; and 4) the MSSVs. The MPS2 control room X/Q values are given in Table 1.3-4.

3.5.4.2 Offsite Atmospheric Dispersion Factors (X/Q)

The EAB and LPZ values used in the CREA analysis have previously been discussed and are listed in Table 1.3-3

3.5.5 CREA Analysis Assumptions and Key Parameters

3.5.5.1 Basic Data and Assumptions

The basic data and assumptions are listed below in Table 3.5.1.

Table 3.5-1 Basic Data and Assumptions for the CREA	
Gap Fractions: (Appendix H of Reference 1)	10% of noble gases and iodines from the core are in the gap
Peaking Factor:	1.69
Control Room Isolated due to a Safety Injection (SI) Signal Initiated after a CREA:	105 seconds
Primary to secondary Leakage Through Each Steam Generator:	150 GPD
Duration of steam releases (time until shutdown cooling is in operation):	60,000 seconds

Table 3.5-1
Basic Data and Assumptions for the CREA

Total Integrated Steam Flow From Both Steam Generators, lbm	Time (sec)	Release (lbm)
	0	0
	2	6,630.11
	6	16,296.40
	24	37,860.48
	26	39,506.87
	30	40,953.53
	70	42,433.26
	100	43,693.41
	200	55,564.74
	600	104,431.90
	650	110,331.41
	800	126,081.07
	1,650	200,833.15
	2,750	288,881.97
	5,000	446,378.59
	6,450	539,179.02
	8,050	640,775.81
	9,000	700,626.29
	10,000	763,624.94
	11,000	826,623.59
	12,000	889,626.65
	12,500	915,949.15
	15,000	1,020,111.65
	20,000	1,220,836.65
	25,000	1,413,911.65
	30,000	1,601,361.65
	35,000	1,784,386.65
	40,000	1,963,761.65
	50,000	2,313,911.65
	60,000	2,654,811.65

3.5.5.2 Millstone Unit 2 Control Room

The dose to the MPS2 control room is based on the CREA described above, the control room assumptions and key values are listed in Tables 1.3-1 and 1.3-2, the X/Qs are listed in Table 1.3-4, and the control room description is given in section 3.1.5.6.

3.5.6 Control Rod Ejection Accident Analysis Results

Table 3.5-2 lists TEDE to the EAB and LPZ from the containment pathway of a CREA at MPS2. Table 3.5-3 lists TEDE to the EAB and LPZ from the secondary system pathway of a CREA at MPS2. The doses to the EAB and LPZ are less than the 6.3 rem TEDE limit stated in 10 CFR 50.67 and RG 1.183. The EAB dose represents the worst 2-hour dose for each release pathway. The dose to the MPS2 control room is less than the 5 rem TEDE limit specified in 10 CFR 50.67 and RG 1.183.

Table 3.5-2
TEDE from a Millstone Unit 2 CREA (containment)

Location	TEDE (rem)
EAB	5.4E-01
LPZ	4.7E-01
Millstone Unit 2 Control Room	1.6E+00

Table 3.5-3
TEDE from a Millstone Unit 2 CREA (secondary side)

Location	TEDE (rem)
EAB	7.9E-01
LPZ	1.8E-01
Millstone Unit 2 Control Room	3.9E+00

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

The NRC, in its rebaselining study of AST impact (Reference 18), considered the effects of the AST on analyses of the postulated integrated radiation doses for plant components exposed to containment atmosphere radiation sources and those exposed to containment sump radiation sources. The NRC study concluded that the increased concentration of cesium in the containment sump water could result in an increase in the postulated integrated doses for certain plant components subject to equipment qualification. The increased cesium concentration in the source term causes (beyond a specific timeframe) the calculated integrated sump doses for the RG 1.183 source term to exceed the doses based upon the TID-14844 source term. The Reference 18 analyses indicated that the timeframe at which the doses based upon the TID-14844 source term may be exceeded and become non-conservative is from approximately 7 to 30 days after the postulated LOCA, depending upon plant-specific assumptions and features.

In the Federal Register notice issuing the final rule for use of alternate source terms at operating reactors (Reference 17), 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 16), 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 re-analyses 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 14), 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 MPS2. The existing equipment qualification analyses, which are based upon the TID-14844 source term, are considered acceptable.

4.2 Risk Impact of Proposed Changes From AST Implementation

Implementation of AST is of benefit to licensees because of the potential to obtain relaxation of specific 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 (accident initiators, malfunctions, dose consequences) as compared with the existing licensing/design bases. The proposed changes associated with implementation of the AST for MPS2 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 2.12:

a. Change Technical Specification Surveillance Requirement 4.7.6.1.e.3, "Control Room Emergency Ventilation System." Verification of the control room in-leakage changes from "less than 130 scfm" to "less than 200 scfm" with the control room emergency ventilation system operating in the recirculation/filtration mode.

   In addition, Actions 3.7.6.1.d and 3.7.6.1.e are stated as applicable during "MODES 5 and 6 and during irradiated fuel movement within containment or the spent fuel pool." This statement will be modified to "MODES 5 and 6 <u>or</u> during irradiated fuel movement within containment or the spent fuel pool."

   This change has minimal risk impact since it cannot initiate an accident or malfunction. It has an impact on control room dose consequences in that a larger in-leakage may result in a higher dose to the operators but dose consequences are within the limits as defined by 10 CFR 50.67. Clarification of the applicability and actions does not change the intent of this technical specification and thus can not increase any risk impact nor can it affect dose consequences.

b. Change Technical Specification Section 3.6.1.2.c "Containment Leakage." The leakage rate acceptance criteria for all penetrations that are secondary containment bypass leakage paths changes from < 0.0072 L_a to < 0.014 L_a.

This change has minimal risk impact since it cannot initiate an accident or malfunction. It has an impact on offsite and control room dose consequences due to increased bypass leakage to the environment but dose consequences are within the limits as defined by 10 CFR 50.67 as well as within guidance provided by RG 1.183.

c. Change Technical Specification 3.4.6.2.c, "Reactor Coolant System Leakage." The current leak rate of 0.035 GPM primary to secondary leakage through any one steam generator is changed from 0.035 GPM to 75 GPD.

   This change has minimal risk impact since it cannot initiate an accident or malfunction. The change was analyzed using a design basis leak rate of 150 GPD to support the proposed technical specification limit of 75 GPD in consideration of Generic Letter 2006-01. It has an impact on offsite and control room dose consequences due to increased primary to secondary leakage to the environment but dose consequences are within the limits as defined by 10 CFR 50.67 as well as within guidance provided by RG 1.183.

d. A new operator action is required to isolate the control room within 4 hours of a main steam line break outside of containment and initiate filtered recirculation consistent with item 4.2.e.

   This change has minimal risk impact since it cannot initiate an accident or malfunction. It has no impact on offsite dose consequences. It has an impact on control room dose consequences in the event of a main steam line break outside containment since the control room may not isolate due to insufficient radioactivity to alarm the control room ventilation radiation monitors. Isolating the control room and initiating filtered recirculation ensures the operators are adequately protected in the event of a main steam line break and dose consequences are within the limits as defined by 10 CFR 50.67.

e. A current operator action to align control room emergency ventilation to filtered recirculation mode within 10 minutes of isolation is revised from 10 minutes to 1 hour.

   This change has minimal risk impact since it cannot initiate an accident or malfunction. It has no impact on offsite dose consequences. It has an impact on control room dose consequences because it delays the onset of filtered recirculation following isolation. Isolating the control room, with subsequent filtered ventilation, ensures the operators are adequately protected in the event of a main steam line break but dose consequences are within the limits as defined by 10 CFR 50.67.

f. Change Technical Specification Section 1.19. The definition of Dose Equivalent 1-131 in Section 1.19 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, " 1988, as the source of thyroid dose conversion factors (Reference 7).

   This change has minimal risk impact since it cannot initiate (or affect) an accident or malfunction. It has an impact on dose consequences only in how they are determined.

g. Change atmospheric dispersion coefficients used in control room dose analyses to reflect use of ARCON96 and RG 1.194.

   This change has minimal risk impact since it cannot initiate an accident or malfunction. It does impact the determination of control room dose consequences as a result of design basis accident analyses.

h. Deletion of Technical Specification LCO 3.3.4 (operability of containment purge valve isolation signal) and all associated surveillance requirements and bases.

   This change has no risk impact since it cannot initiate (or affect) an accident or malfunction. This change has no dose consequence impact because other technical specifications currently require the containment purge valves to be sealed closed without reliance on a containment purge valve isolation signal. Also, radiological accident analyses do not credit this signal. This deletion eliminates the requirement for operability of the containment purge valve isolation signal.

i. Clarification of TS 3/4.3.3, Action b. This change has no risk impact or dose consequences because it only provides clarification to existing wording in an action statement. It does not change the intent of the action statement.

It is concluded that the proposed changes associated with AST implementation for MPS2 will have no significant effect upon the risk associated with the accidents described. 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 changes proposed herein.

4.3 Impact Upon Emergency Planning Dose Assessment Methods

This application of the AST for MPS2 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.

4.4 Impact Upon the Millstone Unit 2 Control Room

The dose consequences to the MPS2 control room operators were assessed for the five accident scenarios considered. Dose consequences were within the requirements of 10 CFR 50.67 and RG 1.183. Operability of the control room boundary and emergency ventilation systems is required.

5.0 Conclusions

The alternate source term defined in RG 1.183 has been incorporated into the reanalysis of radiological dose consequences from five key accidents for MPS2. This amendment request represents a full implementation of the alternate 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.a (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. Dominion Nuclear Connecticut Letter dated May 27, 2004, E. S. Grecheck (Dominion) to USNRC, "Millstone Power Station Unit 3, Proposed Technical Specification Changes, Implementation of Alternate Source Term," Letter Serial No. 04-285, Accession # ML041560464.

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

13. NRC Letter from S. Dembek to M. L. Bowling, dated March 10, 1999, "Issuance of Amendment - Millstone Power Station, Unit 2 (TAC Nos. MA3410 and MA3672)."

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

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

16. 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.

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

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

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

20. NRC Letter dated September 20, 2004, Victor Nerses (USNRC) to D. A. Christian (DNC), "Millstone Power Station Unit No. 2 – Issuance of Amendment Re: Selective Implementation of Alternate Source Term (TAC No. MB6479)"

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

22. NRC 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)."

23. Dominion Nuclear Connecticut Letter dated August 24, 2004, D. A. Christian (Dominion) to USNRC, "Millstone Power Station Unit 2, License Basis Document Change Request LBDCR 2-18-02, Selective Implementation of the Alternate Source Term – Fuel Handling Accident Analyses," Letter Serial No. 04-501, Accession # ML042390046.

24. NRC GENERIC LETTER 2006-01: "Steam Generator Tube Integrity and Associated Technical Specifications," dated January 20, 2006.

7.0 Technical Specification Change

The following technical specifications for MPS2 are revised as noted below to reflect implementation of the RG 1.183 Alternate Source Term (AST) as the design basis source term. The AST implementation analyses provide justification for the following changes to the MPS2 technical specifications:

a. The definition of Dose Equivalent I-131 in Section 1.19 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," 1988, 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 FGR 11 instead of those listed in NRC RG 1.109, Revision 1.

b. Change Technical Specification 3/4.3.3, "Radiation Monitoring," Action 3.3.3.1.b to clarify the wording. The wording "With one or more radiation monitoring channels inoperable" is replaced with "With the number of OPERABLE channels less than the number of minimum channels OPERABLE in Table 3.3.6"

   Table 3.3-6 lists only the number of minimum channels OPERABLE. The number of minimum channels OPERABLE does not exceed the value of 1 for any of the instruments listed in the table. The current wording is inconsistent with information provided in Table 3.3-6 and with the action statement wording provided for ACTIONs 14, 16 and 17 in that:

   1. The wording "One or more" implies that the number of channels required to be OPERABLE could be greater than 1. As indicated above, the number of minimum channels OPERABLE does not exceed the value of 1 for any of the instruments listed in the table.
   2. The action statement does not address the number of channels OPERABLE in relation to the required number of minimum channels OPERABLE.

c. Delete Technical Specification 3/4.3.4, "Containment Purge Valve Isolation Signal." The containment purge valve isolation signal (CPVIS) is not 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 one or more of 10 CFR 50.36(c)(2)(ii) criteria for which technical specifications must be established. This can be justified as follows:

Justification

Based on Technical Specification 3/4.3.4, the CPVIS is required to be OPERABLE in MODES 1 through 4 to ensure the containment purge valves receive the signal to close in the event of an accident to prevent the uncontrolled release of activity to the environment. Technical Specification 3/4.6.3, "Containment Isolation Systems," Limiting Condition For Operation (LCO) 3.6.3.2, has requirements that the containment purge supply and exhaust isolation valves shall be sealed closed while in MODES 1 through 4. This LCO ensures that the purge valves remain closed in MODES 1 through 4 regardless of whether a CPVIS exists or not. The following discussion demonstrates that Technical Specification 3/4.3.4 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.

Technical Specification 3/4.3.4 is specific to the CPVIS, which is a closure signal for the containment purge valves. Since these valves are sealed closed in MODES 1 through 4, a CPVIS cannot change the sealed closed condition of the valves. Technical Specification 3/4.3.4 is unrelated to any detection or indication in the control room of 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 design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Technical Specification 3/4.3.4 provides OPERABILITY requirements

for the CPVIS and 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. The design feature of this signal ensures a closure signal is provided to the containment purge valves in the event they are open. Since they are required to be sealed closed in MODES 1 through 4, the CPVIS serves no purpose, therefore this specification does not satisfy Criterion 2.

Criterion 3
A structure, system, 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.

Technical Specification 3/4.3.4 requires the CPVIS to be operable in MODES 1 through 4. The original intent was to prevent the uncontrolled release of radioactivity to the environment by having a signal capable of closing the purge valves in the event of an accident. Since the purge valves are required to be sealed closed (Technical Specification 3/4.6.3.2), the deletion of OPERABILITY requirements for CPVIS does not challenge or affect the integrity of any fission product barrier and continues to maintain the containment as a fission product barrier, therefore this specification does not satisfy Criterion 3.

Criterion 4
A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The deletion of OPERABILITY requirements for the CPVIS has no impact on public health or safety. The CPVIS is not credited in any radiological accident analysis with mitigation of the consequences of an accident. The CPVIS provides a closure signal to the containment purge valves during MODES 1 through 4 but Technical Specification 3/4.6.3.2 requires that the valves be sealed closed during MODES 1 through 4 anyways. Elimination of the OPERABILITY requirement for the CPVIS has no impact on public health and safety, therefore this specification does not satisfy Criterion 4.

In conclusion, the proposed deletion of Technical Specification 3/4.3.4 does not affect any SSC, 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 deletion of Technical Specification 3/4.3.4 is consistent with regulation and is safe.

d. In Technical Specification 3/4.3.4.6.2, "Reactor Coolant System Leakage," LCO 3.4.6.2.c, the primary to secondary leak rate criterion is changed from 0.035 GPM to 75 GPD. In consideration of Generic Letter 2006-01, the AST implementation analyses, as described in Section 3, assume a primary to secondary leakage rate of 150 GPD to provide a conservative margin above the operational leakage limit.

e. Change Technical Specification 3/4.3.6.1.2, "Containment Leakage," LCO 3.6.1.2.c for all secondary containment bypass leakage paths from a combined leakage rate of $<0.0072\ L_a$ to $<0.014\ L_a$. The AST implementation analyses, as described in section 3, assume leakage rate acceptance criteria for all secondary containment bypass leakage to be $<0.014L_a$.

f. Changes to Technical Specification 3/4.7.6.1, "Control Room Emergency Ventilation System."

   1. ACTION statement on page 3/4 7-16a before ACTIONs d and e by replacing the wording:
      "MODES 5 and 6 and during irradiated fuel movement within containment or the spent fuel pool."
      With
      "MODES 5 and 6 or during irradiated fuel movement within containment or the spent fuel pool."

      This change removes certain ambiguities in the current wording. The use of the logical connector "and" could be interpreted as that Actions are applicable in MODES 5 only when irradiated fuel is moved in the spent fuel pool. Similarly, it could also be interpreted that Actions are applicable in MODES 6 only when irradiated fuel is moved in containment or in the spent fuel pool. The proposed wording defines "the movement of irradiated fuel in the containment or spent fuel pool" as a separate condition in addition to MODES 5 and 6. The proposed wording is consistent with the wording used in NUREG 1432, Rev. 3 (LCO 3.7.11, Action E).

2. The in-leakage rate in Surveillance Requirement 4.7.6.1.e.3 is changed from "less than 130 SCFM" to "less than 200 SCFM". The AST implementation analyses, as described in section 3, assume in-leakage acceptance criteria for all secondary containment bypass leakage to be less than 200 SCFM. The proposed 200 scfm control room habitability envelope inleakage surveillance acceptance criteria has no adverse impact on control room habitability analyses for postulated toxic chemical release events. These habitability analyses do not credit automatic or manual isolation of the control room fresh air ventilation flow during a toxic chemical release event. The control room's forced ventilation fresh air exchange rate (e.g., 800 scfm) is much greater than the proposed 200 scfm envelope inleakage rate acceptance criteria.

Index pages V and XI are revised to reflect the deletion of Technical Specification 3/4.3.4. 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 (Technical Specifications Section 6.23), 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. Applicable Bases changes are included with each technical specification that is changed.

7.1.1 *Definitions*

Revise the current definition of Technical Specification Section 1.19 for DOSE EQUIVALENT I-131 from:

"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 Regulatory Guide 1.109 Rev. 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 I-131 shall be that concentration of I-131 (micro-curie/gram) which alone would produce the same CEDE-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 under inhalation in Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion."

7.1.2 Technical Specification 3/4.3, "Instrumentation"

Revise Radiation Monitoring Limiting Condition for Operation 3.3.3.1.b from:

"Within one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6. The provisions of Specification 3.0.3 are not applicable."

To:

"With the number of OPERABLE channels less than the number of MINIMUM CHANNELS OPERABLE requirement of Table 3.3-6, take the ACTION shown in Table 3.3-6. The provisions of Specification 3.0.3 are not applicable."

7.1.3 Technical Specification 3/4.3.3, "Monitoring Instrumentation"

Delete Technical Specification 3/4.3.4, "Containment Purge Valve Isolation Signal." The text in Technical Specification pages 3/4 3-36 and 3/4 3-37 is replaced with:

"THIS PAGE INTENTIONALLY LEFT BLANK"

7.1.4 Technical Specification 3/4.4.6, "Reactor Coolant System Leakage"

Revise Limiting Condition for Operation 3.4.6.2.c from:

"0.035 GPM primary to secondary leakage through any one steam generator, and"

To:

"75 GPD primary to secondary leakage through any one steam generator, and"

7.1.5 Technical Specification 3/4.6.1, "Primary Containment"

Revise Limiting Condition for Operation 3.6.1.2.c from:

"A combined leakage rate of < 0.0072 L_a for all penetrations that are secondary containment bypass leakage paths when pressurized to P_a."

To:

"A combined leakage rate of < <u>0.014</u> L_a for all penetrations that are secondary containment bypass leakage paths when pressurized to P_a."

7.1.6 Technical Specification 3/4.7.6, "Control Room Emergency Ventilation System"

Revise ACTION from:

"MODES 5 and 6 and during irradiated fuel movement within containment or the spent fuel pool."

To:

"MODES 5 and 6 or during irradiated fuel movement within containment or the spent fuel pool."

Revise Surveillance Requirement 4.7.6.1.e.3 from:

"Verifying that control room air in-leakage is less than 130 SCFM with the Control Room Emergency Ventilation System operating in the recirculation/ filtration mode."

To:

"Verifying that control room air in-leakage is less than 200 SCFM with the Control Room Emergency Ventilation System operating in the recirculation/ filtration mode."

Serial No. 05-662
Docket No. 50-336

**ENCLOSURE 1 TO
ATTACHMENT 1**

**LICENSE BASIS DOCUMENT CHANGE REQUEST (LBDCR) 04-MP2-011
PROPOSED TECHNICAL SPECIFICATION CHANGES
IMPLEMENTATION OF ALTERNATE SOURCE TERM**

ARCON96 INPUTS TO SUPPORT CONTROL ROOM X/Qs

**DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2**

Containment Enclosure Building – Ground Release

Type	Parameter	Value	Units
Input Data File	Input / Run Specification File	*(User Defined)*	
Meteorological Input	Met Data File Name(s)	* MP9701G1.MET	
	Number of Met Data File(s)	1	
	Lower Measurement Height	10.1	m
	Upper Measurement Height	43.3	m
	Wind Speed	m/s	
Receptor Input	Distance to Receptor	20.7	m
	Intake Height	12.3	m
	Elevation Difference	0.0	m
	Direction to Source	345	degrees TN
Source Input	Release Type	Ground	
	Release Height	12.3	m
	Building Area	1087.0	m^2
	Vertical Velocity	0.00	m/s
	Stack Flow	0.00	m^3/s
	Stack Radius	0.00	m
Default Values	Surface Roughness Length	0.20	m
	Wind Direction Window	90	degrees
	Minimum Wind Speed	0.5	m/s
	Averaging Sector Width Constant	4.3	
	Initial Diffusion Coefficient - Sigma Y	7.55	m
	Initial Diffusion Coefficient - Sigma Z	3.86	m
	Hours in Averages	1,2,4,8,12,24,96,168,360,720	hours
	Minimum Number of Hours	1,2,4,8,11,22,87,152,324,648	hours
Output	Output File	*(User Defined)*	
	CFD File	*(User Defined)*	
	Expanded	n	

** Millstone 1997-2001 ground-level met data previously provided in Reference 11*

Refuel Water Storage Tank (RWST) Vent

Type	Parameter	Value	Units
Input Data File	Input / Run Specification File	*(User Defined)*	
Meteorological Input	Met Data File Name(s)	* MP9701G1.MET	
	Number of Met Data File(s)	1	
	Lower Measurement Height	10.1	m
	Upper Measurement Height	43.3	m
	Wind Speed	m/s	
Receptor Input	Distance to Receptor	78.2	m
	Intake Height	12.3	m
	Elevation Difference	0.0	m
	Direction to Source	025	degrees TN
Source Input	Release Type	Ground	
	Release Height	14.3	m
	Building Area	798.0	m^2
	Vertical Velocity	0.00	m/s
	Stack Flow	0.00	m^3/s
	Stack Radius	0.00	m
Default Values	Surface Roughness Length	0.20	m
	Wind Direction Window	90	degrees
	Minimum Wind Speed	0.5	m/s
	Averaging Sector Width Constant	4.3	
	Initial Diffusion Coefficient - Sigma Y	0.00	m
	Initial Diffusion Coefficient - Sigma Z	0.00	m
	Hours in Averages	1,2,4,8,12,24,96,168,360,720	hours
	Minimum Number of Hours	1,2,4,8,11,22,87,152,324,648	hours
Output	Output File	*(User Defined)*	
	CFD File	*(User Defined)*	
	Expanded	n	

** Millstone 1997-2001 ground-level met data previously provided in Reference 11*

Atmospheric Dump Valves (ADVs)

Type	Parameter	Value	Units
Input Data File	Input / Run Specification File	*(User Defined)*	
Meteorological Input	Met Data File Name(s)	* MP9701G1.MET	
	Number of Met Data File(s)	1	
	Lower Measurement Height	10.1	m
	Upper Measurement Height	43.3	m
	Wind Speed	m/s	
Receptor Input	Distance to Receptor	25.1	m
	Intake Height	12.3	m
	Elevation Difference	0.0	m
	Direction to Source	314	degrees TN
Source Input	Release Type	Ground	
	Release Height	23.5	m
	Building Area	1213.3	m^2
	Vertical Velocity	0.00	m/s
	Stack Flow	0.00	m^3/s
	Stack Radius	0.00	m
Default Values	Surface Roughness Length	0.20	m
	Wind Direction Window	90	degrees
	Minimum Wind Speed	0.5	m/s
	Averaging Sector Width Constant	4.3	
	Initial Diffusion Coefficient - Sigma Y	0.00	m
	Initial Diffusion Coefficient - Sigma Z	0.00	m
	Hours in Averages	1,2,4,8,12,24,96,168,360,720	hours
	Minimum Number of Hours	1,2,4,8,11,22,87,152,324,648	hours
Output	Output File	*(User Defined)*	
	CFD File	*(User Defined)*	
	Expanded	n	

** Millstone 1997-2001 ground-level met data previously provided in Reference 11*

Turbine Building Blowout Panels

Type	Parameter	Value	Units
Input Data File	Input / Run Specification File	*(User Defined)*	
Meteorological Input	Met Data File Name(s)	* MP9701G1.MET	
	Number of Met Data File(s)	1	
	Lower Measurement Height	10.1	m
	Upper Measurement Height	43.3	m
	Wind Speed	m/s	
Receptor Input	Distance to Receptor	19.5	m
	Intake Height	12.3	m
	Elevation Difference	0.0	m
	Direction to Source	278	degrees TN
Source Input	Release Type	Ground	
	Release Height	21.9	m
	Building Area	693.6	m^2
	Vertical Velocity	0.00	m/s
	Stack Flow	0.00	m^3/s
	Stack Radius	0.00	m
Default Values	Surface Roughness Length	0.20	m
	Wind Direction Window	90	degrees
	Minimum Wind Speed	0.5	m/s
	Averaging Sector Width Constant	4.3	
	Initial Diffusion Coefficient - Sigma Y	0.00	m
	Initial Diffusion Coefficient - Sigma Z	0.00	m
	Hours in Averages	1,2,4,8,12,24,96,168,360,720	hours
	Minimum Number of Hours	1,2,4,8,11,22,87,152,324,648	hours
Output	Output File	*(User Defined)*	
	CFD File	*(User Defined)*	
	Expanded	n	

** Millstone 1997-2001 ground-level met data previously provided in Reference 11*

Main Steam Safety Valves (MSSVs)

Type	Parameter	Value	Units
Input Data File	Input / Run Specification File	*(User Defined)*	
Meteorological Input	Met Data File Name(s)	* MP9701G1.MET	
	Number of Met Data File(s)	1	
	Lower Measurement Height	10.1	m
	Upper Measurement Height	43.3	m
	Wind Speed	m/s	
Receptor Input	Distance to Receptor	29.9	m
	Intake Height	12.3	m
	Elevation Difference	0.0	m
	Direction to Source	325	degrees TN
Source Input	Release Type	Ground	
	Release Height	44.5	m
	Building Area	1087.0	m^2
	Vertical Velocity	0.00	m/s
	Stack Flow	0.00	m^3/s
	Stack Radius	0.00	m
Default Values	Surface Roughness Length	0.20	m
	Wind Direction Window	90	degrees
	Minimum Wind Speed	0.5	m/s
	Averaging Sector Width Constant	4.3	
	Initial Diffusion Coefficient - Sigma Y	0.00	m
	Initial Diffusion Coefficient - Sigma Z	0.00	m
	Hours in Averages	1,2,4,8,12,24,96,168,360,720	hours
	Minimum Number of Hours	1,2,4,8,11,22,87,152,324,648	hours
Output	Output File	*(User Defined)*	
	CFD File	*(User Defined)*	
	Expanded	n	

** Millstone 1997-2001 ground-level met data previously provided in Reference 11*

Serial No. 05-662
Docket No. 50-336

ATTACHMENT 2

LICENSE BASIS DOCUMENT CHANGE REQUEST (LBDCR) 04-MP2-011
PROPOSED TECHNICAL SPECIFICATION CHANGES
IMPLEMENTATION OF ALTERNATE SOURCE TERM

MARKED UP TECHNICAL SPECIFICATION PAGES

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

AZIMUTHAL POWER TILT - T_q

1.18 AZIMUTHAL POWER TILT shall be the difference between the maximum power generated in any core quadrant (upper or lower) and the average power of all quadrants in that half (upper or lower) of the core divided by the average power of all quadrants in that half (upper or lower) of the core.

$$\text{AZIMUTHALPOWERTILT} = \left[\frac{\text{Maximum power in any core quadrant (upper or lower)}}{\text{Average power of all quadrants (upper or lower)}}\right] - 1$$

DOSE EQUIVALENT I-131 CEDE-thyroid

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (micro-curie/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 Regulatory Guide 1.109 REv. 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."~~ Insert A

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.20 $\bar{E}$ shall be the average sum of the beta and gamma energies per disintegration (in MEV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

STAGGERED TEST BASIS

1.21 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subinterval, and

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

FREQUENCY NOTATION

1.22 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

Insert A to Page 1-4

under inhalation in Federal Guidance Report No. 11 (FGR11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion."

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 2 hours or declare the channel inoperable.

b. ~~With one or more radiation monitoring channels inoperable,~~ take the ACTION shown in Table 3.3-6. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.

4.3.3.1.2 DELETED

4.3.3.1.3 Verify the response time of the control room isolation channel at least once per 18 months.

With the number of OPERABLE channels less than the number of MINIMUM CHANNELS OPERABLE in Table 3.3-6

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Deleted					
b. Control Room Isolation	1	ALL MODES	2 mR/hr	10^{-1} - 10^{4} mR/hr	16
c. Containment High Range	1	1,2,3,&4	100 R/hr	10^{0} - 10^{8} R/hr	17
2. PROCESS MONITORS					
a. Containment Atmosphere-Particulate	1	1, 2, 3, & 4	NA	10 - 10^{+6} cpm	14
b. Containment Atmosphere-Gaseous	1	1, 2, 3, & 4	NA	10 - 10^{+6} cpm	14
c. Noble Gas Effluent Monitor (high range) (Unit 2 stack)	1	1, 2, 3, & 4	2 x 10^{-1} uci/cc	10^{-3} - 10^{5} uci/cc	17

For Information Only

TABLE 3.3-6 (Continued)

TABLE NOTATION

For Information Only

(a) DELETED

ACTION 13 - DELETED

ACTION 14 - With the number of process monitors OPERABLE less than required by the MINIMUM CHANNELS OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 15 - DELETED

ACTION 16 - With the number of OPERABLE channels less than required by the MINIMUM CHANNELS OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

ACTION 17 - With the number of OPERABLE channels less than required by the MINIMUM CHANNELS OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) either restore the inoperable channel(s) to OPERABLE status within 7 days of the discovery or

2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following discovery outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3-3
RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT			CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1.	AREA MONITORS					
	a.	Deleted				
	b.	Control Room Isolation	S	R	M	ALL MODES
	c.	Containment High Range	S	R*	M	1, 2, 3, & 4
2.	PROCESS MONITORS					
	a.	Containment Atmosphere-Particulate	S	R	M	1, 2, 3, & 4
	b.	Containment Atmosphere-Gaseous	S	R	M	1, 2, 3, & 4
	c.	Noble Gas Effluent Monitor (high range) (Unit 2 Stack)	S	R	M	1, 2, 3, & 4

For Information Only

* Calibration of the sensor with a radioactive source need only be performed on the lowest range. Higher ranges may be calibrated electronically.

INSTRUMENTATION

CONTAINMENT PURGE VALVE ISOLATION SIGNAL

LIMITING CONDITION FOR OPERATION

3.3.4 One Containment Purge Valve Isolation Signal containment gaseous radiation monitor channel, one Containment Purge Valve Isolation Signal containment particulate radiation monitor channel, and one Containment Purge Valve Isolation Signal automation logic train shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With no OPERABLE containment purge valve isolation signal, containment gaseous radiation monitor channel, containment purge valve isolation signal, containment particulate radiation monitor channel, and containment purge valve isolation signal automatic logic train, enter the applicable conditions and required ACTIONS for the affected valves of Technical Specification 3.6.3.1, "Containment Isolation Valves."

SURVEILLANCE REQUIREMENTS

4.3.4.1 Perform a CHANNEL CHECK on each Containment Purge Valve Isolation Signal containment gaseous and particulate radiation monitor channel at least once per 12 hours.

4.3.4.2 Perform a CHANNEL FUNCTIONAL TEST on each Containment Purge Valve Isolation Signal containment gaseous and particulate radiation monitor channel at least once per 31 days.

This surveillance shall include verification of the trip value in accordance with the following:

The trip value shall be such that the containment purge effluent shall not result in calculated concentrations of radioactivity offsite in excess of 10 CFR Part 20, Appendix B, Table II. For the purposes of calculating this trip value, a $x/Q = 5.8 \times 10^{-6}$ sec/m^3 shall be used when the system is aligned to purge through the building vent and a $x/Q = 7.5 \times 10^{-8}$ sec/m^3 shall be used when the system is aligned to purge through the Unit 1 stack, the gaseous and particulate (Half Lives greater than 8 days) radioactivity shall be assumed to be Xe-133 and Cs-137, respectively. However, the setpoints shall be no greater than 5×10^5 cpm.

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SURVEILLANCE REQUIREMENTS

4.3.4.3 Perform a CHANNEL FUNCTIONAL TEST on each Containment Purge Valve Isolation Signal automatic actuation logic train at least once per 31 days. This actuation logic shall include verification of the proper operation of the actuation relays.

4.3.4.4 Perform a CHANNEL CALIBRATION on each Containment Purge Valve Isolation Signal containment gaseous and particulate radiation monitor channel at least once per 18 months.

4.3.4.5 Verify Containment Purge Valve Isolation Signal response time at least once per 18 months. Each test shall include at least one containment gaseous and one containment particulate radiation monitor channel such that all channels are tested at least once every N times 18 months where N is the total number of containment gaseous or total number of containment particulate radiation monitor channels.

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REACTOR COOLANT SYSTEM LEAKAGE

~~March 10, 1999~~

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 75 GPD ~~0.035 GPM~~ primary-to-secondary leakage through any one steam generator, and

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in COLD SHUTDOWN within 36 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System IDENTIFIED LEAKAGE and UNIDENTIFIED LEAKAGE shall be demonstrated to be within limits by performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation except when operating in the shutdown cooling mode.

4.4.6.2.2 Primary to secondary leakage shall be demonstrated to be within the above limits by performance of a primary to secondary leak rate determination at least once per 72 hours. The provisions of Specification 4.0.4 are not applicable for entry into MODE 4.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of < L_a, 0.50 percent by weight of the containment air per 24 hours at P_a, 54 psig.

b. A combined leakage rate of < 0.60 L_a for all penetrations and valves subject to Type B and C tests when pressurized to P_a.

c. A combined leakage rate of < ~~0.0072~~ 0.014 L_a for all penetrations that are secondary containment bypass leakage paths when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, or (c) with the combined bypass leakage rate exceeding ~~0.0072~~ 0.014 L_a, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated in accordance with the Containment Leakage Rate Testing Program.

For Information Only

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 Two independent Control Room Emergency Ventilation Trains shall be OPERABLE.*

APPLICABILITY: MODES 1, 2, 3, 4, 5 and 6.

During irradiated fuel movement within containment or the spent fuel pool.

ACTION:

MODES 1, 2, 3, and 4:

a. With one Control Room Emergency Ventilation Train inoperable, restore the inoperable train 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 Ventilation Trains inoperable, except as specified in ACTION c., immediately suspend the movement of irradiated fuel assemblies within the spent fuel pool. Restore at least one inoperable train 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 Ventilation Trains inoperable due to an inoperable Control Room boundary, immediately suspend the movement of irradiated 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.

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

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

ACTION (continued)

MODES 5 and 6, or ~~and~~ during irradiated fuel movement within containment or the spent fuel pool:**

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

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

** In MODES 5 and 6, when a Control Room Emergency Ventilation Train is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of 3.7.6.1 Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system (s), subsystem (s), train (s), component (s) and device(s) are OPERABLE, or likewise satisfy the requirements of the specification. Unless both conditions (1) and (2) are satisfied within 2 hours, then ACTION 3.7.6.1.d or 3.7.6.1.e shall be invoked as applicable.

For Information Only

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For Information Only

4.7.6.1 Each Control Room Emergency Ventilation Train shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is ≤ 100°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 absorber train and verifying that the train operates for at least 15 minutes.

c. At least once per 18 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 train by:

   1. Verifying that the cleanup train satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the train flow rate is 2500 cfm ± 10%.

   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, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.* The carbon sample shall have a removal efficiency of ≥ 95 percent.

   3. Verifying a train flow rate of 2500 cfm ± 10% during train operation when tested in accordance with ANSI N510-1975.

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, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

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* ASTM D3803-89 shall be used in place of ANSI N509-1976 as referenced in table 2 of Regulatory Guide 1.52. The laboratory test of charcoal should be conducted at a temperature of 30°C and a relative humidity of 95% within the tolerances specified by ASTM D3803-89.

For Information Only

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 3.4 inches Water Gauge while operating the train at a flow rate of 2500 cfm ± 10%.

2. Verifying that on a recirculation signal, with the Control Room Emergency Ventilation Train operating in the normal mode and the smoke purge mode, the train automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that control room air in-leakage is less than 200 ~~130~~ SCFM with the Control Room Emergency Ventilation System operating in the recirculation/filtration mode.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 2500 cfm ± 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 2500 cfm ± 10%.

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Docket No. 50-336

ATTACHMENT 3

LICENSE BASIS DOCUMENT CHANGE REQUEST (LBDCR) 04-MP2-011
PROPOSED TECHNICAL SPECIFICATION CHANGES
IMPLEMENTATION OF ALTERNATE SOURCE TERM

RETYPED TECHNICAL SPECIFICATION PAGES

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION PAGE

AZIMUTHAL POWER TILT - T_q

1.18 AZIMUTHAL POWER TILT shall be the difference between the maximum power generated in any core quadrant (upper or lower) and the average power of all quadrants in that half (upper or lower) of the core divided by the average power of all quadrants in that half (upper or lower) of the core.

$$\text{AZIMUTHALPOWERTILT} = \left[\frac{\text{Maximum power in any core quadrant (upper or lower)}}{\text{Average power of all quadrants (upper or lower)}}\right] - 1$$

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (micro-curie/gram) which alone would produce the same CEDE-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 under inhalation in Federal Guidance Report No. 11 (FGR11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion."

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.20 $\bar{E}$ shall be the average sum of the beta and gamma energies per disintegration (in MEV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

STAGGERED TEST BASIS

1.21 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subinterval, and

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

FREQUENCY NOTATION

1.22 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 2 hours or declare the channel inoperable.

b. With the number of OPERABLE channels less than the number of MINIMUM CHANNELS OPERABLE in Table 3.3-6, take the ACTION shown in Table 3.3-6. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.

4.3.3.1.2 DELETED

4.3.3.1.3 Verify the response time of the control room isolation channel at least once per 18 months.

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REACTOR COOLANT SYSTEM LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 75 GPD primary-to-secondary leakage through any one steam generator, and

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in COLD SHUTDOWN within 36 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System IDENTIFIED LEAKAGE and UNIDENTIFIED LEAKAGE shall be demonstrated to be within limits by performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation except when operating in the shutdown cooling mode.

4.4.6.2.2 Primary to secondary leakage shall be demonstrated to be within the above limits by performance of a primary to secondary leak rate determination at least once per 72 hours. The provisions of Specification 4.0.4 are not applicable for entry into MODE 4.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of $< L_a$, 0.50 percent by weight of the containment air per 24 hours at P_a, 54 psig.

b. A combined leakage rate of $< 0.60\ L_a$ for all penetrations and valves subject to Type B and C tests when pressurized to P_a.

c. A combined leakage rate of $< 0.014\ L_a$ for all penetrations that are secondary containment bypass leakage paths when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding $0.75\ L_a$, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding $0.60\ L_a$, or (c) with the combined bypass leakage rate exceeding $0.014\ L_a$, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated in accordance with the Containment Leakage Rate Testing Program.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

ACTION (continued)

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

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

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

** In MODES 5 and 6, when a Control Room Emergency Ventilation Train is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of 3.7.6.1 Limiting Condition for Operation, provided: (1) its corresponding normal <u>or</u> emergency power source is OPERABLE; and (2) all of its redundant system (s), subsystem (s), train (s), component (s) and device(s) are OPERABLE, or likewise satisfy the requirements of the specification. Unless both conditions (1) and (2) are satisfied within 2 hours, then ACTION 3.7.6.1.d or 3.7.6.1.e shall be invoked as applicable.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that control room air in-leakage is less than 200 SCFM with the Control Room Emergency Ventilation System operating in the recirculation/filtration mode.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 2500 cfm ± 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 2500 cfm ± 10%.

ATTACHMENT 4

LICENSE BASIS DOCUMENT CHANGE REQUEST (LBDCR) 04-MP2-011
PROPOSED TECHNICAL SPECIFICATION CHANGES
IMPLEMENTATION OF ALTERNATE SOURCE TERM

SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

This license amendment proposes full implementation of an alternative source term (AST) and changes to the Millstone Power Station Unit 2 (MPS2) 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 3/4.3.3, Radiation Monitoring – revised Action 3.3.3.1.b to clarify the wording.
- Technical Specification 3/4.3.4, Containment Purge Valve Isolation Signal – deleted.
- Technical Specification 3/4.3.4.6.2, Reactor Coolant System Leakage - changed the value used for primary to secondary leak rate acceptance criteria.
- Technical Specification 3/4.3.6.1.2, Containment Leakage, LCO 3.6.1.2.c – changed the value used for all secondary containment bypass leakage paths.
- Technical Specification 3/4.7.6.1, Control Room Emergency Ventilation System – clarified Action statement.
- Surveillance Requirement 4.7.6.1.e.3 - changed the value used in the acceptance criteria of in-leakage rate.

Dominion Nuclear Connecticut, Inc. has 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, will operation of Millstone Power Station Unit 2 with the proposed changes:

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

   No.

   The proposed amendment does not involve a significant increase in the probability or consequence of an accident previously analyzed. The MPS2 control room emergency ventilation system only functions following the initiation of a design basis radiological accident. Therefore, the change to the value used for in-leakage rate test acceptance criteria following a design basis accident will not increase the probability of any previously analyzed accident. The proposed 200 cfm control room habitability envelope inleakage surveillance acceptance criteria has no adverse impact on control room habitability analyses for postulated toxic chemical release events. These habitability analyses do not credit automatic or manual isolation of the control room fresh air ventilation flow during a toxic chemical release event. The control room's forced ventilation fresh air exchange rate (e.g., 800 cfm) is much greater than the proposed 200 cfm envelope inleakage rate acceptance criteria. The MPS2 containment purge valve isolation signal is not

credited in the accident 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 any previously analyzed accident. The MPS2 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 primary to secondary leak rate acceptance criteria, and for all secondary containment bypass leakage paths following a design basis accident, will not increase the probability of any previously analyzed accident.

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 MPS2 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?

   No.

   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 ventilation system, and the containment and 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?

   No.

   The implementation of the proposed changes does not reduce the margin of safety. The proposed changes for the control room ventilation system, and the containment and 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 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

LICENSE BASIS DOCUMENT CHANGE REQUEST (LBDCR) 04-MP2-011
PROPOSED TECHNICAL SPECIFICATION CHANGES
IMPLEMENTATION OF ALTERNATE SOURCE TERM

ENVIRONMENTAL IMPACT EVALUATON

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

ENVIRONMENTAL IMPACT EVALUATON

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 Connecticut, 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 MPS2 design basis radiological accidents. Approval of a new alternative source term for MPS2 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 ventilation system during radiological accidents, 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 changes 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 operator actions 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 changes and implementation of an alternative source term will not result in a significant increase in individual or cumulative occupational radiation exposure.

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ATTACHMENT 6

LICENSE BASIS DOCUMENT CHANGE REQUEST (LBDCR) 04-MP2-011
PROPOSED TECHNICAL SPECIFICATION CHANGES
IMPLEMENTATION OF ALTERNATE SOURCE TERM

MARKED UP TECHNICAL SPECIFICATION BASES PAGES
(FOR INFORMATION ONLY)

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

BASES(Continued)

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

The analyses for a Steam Generator Tube Rupture, ~~Event and for a Millstone Unit No. 3~~ Loss of Coolant Accident, credit the control room ventilation inlet duct radiation monitors with closure of the Unit 2 control room isolation dampers. In the event of a single failure in either channel (1 per train), the control room isolation dampers automatically close. The response time test for the control room isolation dampers includes signal generation time and damper closure. The response time for the control room isolation dampers is maintained within the applicable facility surveillance procedure.

The containment airborne radiation monitors (gaseous and particulate) provide early indication of leakage from the Reactor Coolant System as specified in Technical Specification 3.4.6.1.

, Rod Ejection Accident and Fuel Handling Accident

3/4.3.3.9 - DELETED

3/4.3.3.10 - DELETED

DELETED

3/4.3.4 ~~Containment Purge Valve Isolation Signal~~

A high airborne radioactivity level inside containment will be detected by the containment airborne radiation monitors (gaseous and particulate). The actuation logic for this function is one out of four. High radioactivity inside containment, detected by any one of the four radiation detectors (two gaseous and two particulate), will automatically isolate containment PURGE.

An OPERABLE system consists of at least one gaseous and particulate radiation detector and the associated automatic logic train. An actuation logic train consists of the detectors, associated microprocessors, and the associated logic circuits up to and including the Engineered Safeguards Actuation System system actuation module.

These radiation monitors provide an automatic closure signal to the containment purge valves upon detection of high airborne radioactivity levels inside containment. The maximum allowable trip value for these monitors corresponds to calculated concentrations at the site boundary which would not exceed the concentrations listed in 10 CFR Part 20, Appendix B, Table II. Exposure for a year to the concentrations in 10 CFR Part 20, Appendix B, Table II, corresponds to a total body dose to an individual of 500 mrem, which is well below the guidelines of 10 CFR Part 100 for an individual at any point on the exclusion area boundary for two hours.

Determination of the monitor's trip value in counts per minute, which is the actual instrument response, involves several factors including: 1) the atmospheric dispersion (x/Q), 2) isotopic composition of the sample, 3) sample flow rate, 4) sample collection efficiency, 5) counting efficiency, and 6) the background radiation level at the detector. The x/Q of 5.8 x 10-6 sec/m3 is the highest annual average x/Q estimated for the site boundary (0.48 miles in the NE sector) for vent releases from the containment and 7.5 x 10-8 sec/m3 is the highest annual average x/Q estimated for an off-site location (3 miles in the NNE sector) for releases from the Unit 1 stack. This calculation also assumes that the isotopic composition is xenon-133 for gaseous radioactivity and cesium-137 for particulate radioactivity (Half Lives greater than 8 days). The upper limit of 5 x 105 cpm is approximately 90 percent of full instrument scale.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

3/4.4.6.2 REACTOR COOLANT SYSTEM LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

75 GPD

The steam generator tube leakage limit of ~~0.035 GPM~~ per steam generator ensures that the dosage contribution from the tube leakage will be less than the limits of ~~General Design Criteria 19 of 10CFR50 Appendix A~~ in the event of either a steam generator tube rupture or steam line break. The ~~0.035 GPM~~ limit is consistent with the assumptions used in the analysis of these accidents.

10 CFR 50.67

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

The IDENTIFIED LEAKAGE and UNIDENTIFIED LEAKAGE limits listed in LCO 3.4.6.2 only apply to the reactor coolant system pressure boundary within the containment.

In accordance with 10 CFR 50.2 "Definitions" the RCS Pressure Boundary means all those pressure-containing components such as pressure vessels, piping, pumps and valves which are (1) Part of the Reactor Coolant System, or (2) Connected to the Reactor Coolant System, up to and including any and all of the following: (i) The outermost containment isolation valve in system piping which penetrates primary reactor containment, (ii) The second of two valves normally closed in system piping which does not penetrate primary reactor containment, or (iii) The reactor coolant safety and relief valves.

The definitions for IDENTIFIED LEAKAGE and UNIDENTIFIED LEAKAGE are provided in the Technical Specifications definitions section, definitions 1.14 and 1.15 respectively.

Leakage outside of the second isolation valve for containment which is included in the RCS Leak Rate Calculation is not considered RCS leakage and can be subtracted from RCS UNIDENTIFIED LEAKAGE.

The safety significance of RCS leakage varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS leakage into the containment area is necessary. Quickly separating IDENTIFIED LEAKAGE from the UNIDENTIFIED LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur. LCO 3.4.6.2 deals with protection of the reactor coolant pressure boundary from degradation and the core from inadequate cooling, in addition accident analysis radiation release assumptions from being exceeded.

REACTOR COOLANT SYSTEM

BASES

3/4.4.7 DELETE

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the SITE BOUNDARY ~~will not exceed an appropriately small fraction of Part 100 limits~~ following a steam generator tube rupture accident.

(1)

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 uCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

are within the dose guidelines of 10 CFR 50.67

3/4.7.4 SERVICE WATER SYSTEM (Continued)

determined to be inoperable should be the loop that results in the most adverse plant configuration with respect to the availability of accident mitigation equipment. Restoration of loop independence within the time constraints of the allowed outage time is required, or a plant shutdown is necessary.

It is acceptable to operate with the SW header supply valves to sodium hypochlorite (2-SW-84A and 2-SW-84B) and the SW header supply valves to the north and south filters(2-SW-298 and 2-SW-299) open.The flow restricting orifices in these lines ensure that safety related loads continue to receive minimum required flow during a LOCA (in which the lines remain intact) or during a seismic event (when the lines break). Therefore, operation with these valves open does not affect OPERABILITY of the SW loops.

Surveillance Requirement 4.7.4.1.a verifies the correct alignment for manual, power operated, and automatic valves in the Service Water (SW) System flow paths to provide assurance that the proper flow paths exist for SW operation. This surveillance does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve automatically repositions within the proper stroke time. This surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day frequency is appropriate because the valves are operated under procedural control and an improper valve position would only affect a single train. This frequency has been shown to be acceptable through operating experience.

Surveillance Requirements 4.7.4.1.b and 4.7.4.1.c demonstrate that each automatic SW valve actuates to the required position on an actual or simulated actuation signal and that each SW pump starts on receipt of an actual or simulated actuation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month frequency is based on the need to perform these surveillances under the conditions that apply during a plant outage and the potential for unplanned transients if the surveillances were performed with the reactor at power. The 18 month frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of the Engineered Safety Feature Actuation System (ESFAS) testing, and equipment performance is monitored as part of the Inservice Testing Program.

3/4.7.5 DELETED

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

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.

PLANT SYSTEMS

BASES

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

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 consistent with the requirements of General Design Criteria 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

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.

The control room radiological dose calculations use the conservative minimum acceptable flow of 2250 cfm based on the flowrate surveillance requirement of 2500 cfm ± 10%.

The LCO is applicable in MODES 1, 2, 3, 4, 5 and 6. It is also applicable during irradiated fuel movement within containment or the spent fuel pool irrespective of any plant MODE.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

Currently there are some situations where the CREV System may not automatically start on an accident signal, without operator action. Under most situations, the emergency filtration fans will start and the CREV System will be in the accident lineup. However, a failure of a supply fan (F21A or B) or an exhaust fan (F31A or B), operator action will be required to return to a full train lineup. Also, if a single emergency bus does not power up for one train of the CREV System, the opposite train filter fan will automatically start, but the required supply and exhaust fans will not automatically start. Therefore, operator action is required to establish the whole train lineup. This action is specified in the Emergency Operating Procedures. The radiological dose calculations do not take credit for CREV System cleanup action until ~~10 minutes~~ 1 hour into the accident to allow for operator action.

When the CREV System is checked to shift to the recirculation mode of operation, this will be performed from the normal mode of operation, and from the smoke purge mode of operation.

With both control room emergency ventilation trains inoperable due to an inoperable control room boundary, the movement of irradiated fuel assemblies 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 ventilation trains 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 into 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.

Surveillance Requirement 4.7.6.1.c.1 dictates the test frequency, methods and acceptance criteria for the Control Room Emergency ventilation System trains (cleanup trains). These criteria all originate in the Regulatory Position sections of Regulatory Guide 1.52, Rev. 2, March 1978 as discussed below.

Section C.5.a requires a visual inspection of the cleanup system be made before the following tests, in accordance with the provisions of section 5 of ANSI N510-1975:

- in-place air flow distribution test
- DOP test
- activated carbon adsorber section leak test

For Information Only

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

Section C.5.c requires the in-place Dioctyl phthalate (DOP) test for HEPA filters to conform to section 10 of ANSI N510-1975. The HEPA filters should be tested in place (1) initially, (2) at least once per 18 months thereafter, and (3) following painting, fire, or chemical release in any ventilation zone communicating with the system. The testing is to confirm a penetration of less than 0.05%* at rated flow. A filtration system satisfying this criteria can be considered to warrant a 99% removal efficiency for particulates.

Section C.5.d requires the charcoal adsorber section to be leak tested with a gaseous halogenated hydrocarbon refrigerant, in accordance with section 12 of ANSI N510-1975 to ensure that bypass leakage through the adsorber section is less than 0.05%.** Adsorber leak testing should be conducted (1) initially, (2) at least once per 18 months thereafter, (3) following removal of an adsorber sample for laboratory testing if the integrity of the adsorber section is affected, and (4) following painting, fire, or chemical release in any ventilation zone communicating with the system.

The ACTION requirements to immediately suspend various activities (CORE ALTERATIONS, irradiated fuel movement, etc.) do not preclude completion of the movement of a component to a safe position.

Technical Specification 3.7.6.1 provides the OPERABILITY requirements for the Control Room Emergency Ventilation Trains. If a Control Room Emergency Ventilation Train emergency power source or normal power source becomes inoperable in MODES 1, 2, 3, or 4 the requirements of Technical Specification 3.0.5 apply in determining the OPERABILITY of the affected Control Room Emergency Ventilation Train. If a Control Room Emergency Ventilation Train emergency power source or normal power source becomes inoperable in MODES 5 or 6 the guidance provided by Note "**" of this specification applies in determining the OPERABILITY of the affected Control Room Emergency Ventilation Train. If a Control Room Emergency Ventilation Train emergency power source or normal power source becomes inoperable while not in MODES 1, 2, 3, 4, 5, or 6 the requirements of Technical Specification 3.0.5 apply in determining the OPERABILITY of the affected Control Room Emergency Ventilation Train.

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* Means that the HEPA filter will allow passage of less than 0.05% of the test concentration injection at the filter inlet from a standard DOP concentration injection.

** Means that the charcoal adsorber sections will allow passage of less than 0.05% of the injected test concentration around the charcoal adsorber section.