May 31, 2007

Mr. David A. Christian
Sr. Vice President and Chief Nuclear Officer
Dominion Nuclear Connecticut, Inc.
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION, UNIT NO.2 - ISSUANCE OF AMENDMENT REGARDING ALTERNATE SOURCE TERM (TAC NO. MD2346)

Dear Mr. Christian:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 298 to Facility Operating License No. DPR-65 for the Millstone Power Station, Unit No. 2 (MPS2), in response to your application dated June 13, 2006, as supplemented by letter dated March 6, 2007.

This amendment revises the MPS2 licensing basis in the area of radiological dose analysis for design-basis accidents using the alternative source term permitted by Title of the *Code of Federal Regulations* 50.67, "Accident source term". Additionally, the amendment revises the MPS2 Technical Specifications to be consistent with the amended licensing-basis.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/ra/

John Hughey, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures:
1. Amendment No. 298 to License No. DPR-65
2. Safety Evaluation

cc w/encls: See next page

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John Hughey, Project Manager
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DATE	5/31/07	5/31/07	05/30/07	05/03/07	05/18/07

OFFICE	DCI/CSGB/BC	DIRS/IOLB/BC	OGC (NLO)	LPLI-2/BC
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DOMINION NUCLEAR CONNECTICUT, INC.

DOCKET NO. 50-336

MILLSTONE POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 298
License No. DPR-65

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

   A. The application for amendment filed by Dominion Nuclear Connecticut, Inc. (the licensee) dated June 13, 2006, as supplemented by letter dated March 6, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-65 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No.   , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. Dominion Nuclear Connecticut, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to License No. DPR-65
and the Technical Specifications

Date of Issuance: May 31, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 298

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

Replace page 3 of License No. DPR-65 with the attached revised page 3.

Replace the following pages of the Appendix A, Technical Specifications with the attached revised pages as indicated. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
V	V
XI	XI
1-4	1-4
3/4 3-24	3/4 3-24
3/4 3-36	3/4 3-36
3/4 3-37	3/4 3-37
3/4 4-9	3/4 4-9
3/4 6-2	3/4 6-2
3/4 7-16a	3/4 7-16a
3/4 7-18	3/4 7-18

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 298

TO FACILITY OPERATING LICENSE NO. DPR-65

DOMINION NUCLEAR CONNECTICUT, INC.

MILLSTONE POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

1.0 INTRODUCTION

By letter dated June 13, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML061940105), as supplemented by letter dated March 6, 2007 (ADAMS Accession No. ML070680039), Dominion Nuclear Connecticut, Inc. (DNC or the licensee) submitted a license amendment request (LAR) for Millstone Power Station, Unit No. 2 (MPS2). This amendment would revise the MPS2 licensing basis in the area of radiological dose analysis for design-basis accidents (DBAs) using the alternative source term permitted by Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50.67, "Accident Source Term," (AST). Additionally, the amendment revises the MPS2 Technical Specifications to be consistent with the amended licensing-basis.

The supplement dated March 6, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards determination as published in the *Federal Register* on August 29, 2006 (71 FR 51226).

2.0 REGULATORY EVALUATION

The U.S. Nuclear Regulatory Commission (NRC) staff evaluated the radiological consequences of affected DBAs for implementation of the AST methodology at MPS2, as proposed by the licensee against the dose criteria specified in 10 CFR Section 50.67(b)(2). These criteria are 25 roentgen equivalent man (rem) total effective dose equivalent (TEDE) at the exclusion area boundary (EAB) for any 2-hour period following the onset of the postulated fission product release, 25 rem TEDE at the outer boundary of the low population zone (LPZ) for the duration of the postulated fission product release, and 5 rem TEDE in the control room (CR) for the duration of the postulated fission product release.

This safety evaluation (SE) addresses the impact of the proposed changes on previously analyzed DBA radiological consequences and the acceptability of the revised analysis results. The regulatory requirements on which the NRC staff based its acceptance are the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 of Regulatory Guide (RG) 1.183, Standard Review Plan (SRP) 15.0.1, and General Design Criterion (GDC) 19. The licensee has not proposed any significant deviation or departure from the guidance provided in RG 1.183. The NRC staff's evaluation is based upon the following regulatory codes, guides, and standards:

- 10 CFR Part 50.67, “Accident Source Term”
- 10 CFR Part 50, Appendix A, "General Design Criterion for Nuclear Power Plants”: GDC 19, “Control room”
- RG 1.23, “Onsite Meteorological Programs”
- RG 1.52, “Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption units of Light-Water-Cooled Nuclear Power Plants”
- RG 1.78, “Evaluating The Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release”
- RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors”
- RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants”
- NUREG-0800, “Standard Review Plan,” Section 6.4, “Control Room Habitability Systems”
- SRP, Section 15.6.2, “Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment”
- SRP Section 2.3.4, “Short-Term Diffusion Estimates for Accidental Atmospheric Releases”
- SRP Section 15.0.1, “Radiological Consequence Analyses Using Alternative Source Terms”

With regard to the proposed changes to manual operator actions, the staff used the guidance contained in NRC Information Notice (IN) 97-78, "Crediting Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times;" ANSI/ANS 58.8-1994, "Time Response Design Criteria for Safety-Related Operator Actions;" and NUREG-0800, "Standard Review Plan," Chapter 18.0, "Human Factors Engineering."

3.0 TECHNICAL EVALUATION

3.1 Radiological Consequences of DBAs

The current licensing basis at MPS2 includes the following DBAs analyzed for radiological consequences at the EAB, LPZ, and in the CR in Section 15, “Accident Analyses” of the MPS2 Updated Final Safety Analysis Report (UFSAR):

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 licensee proposed relocating the radiological consequences of the WGSF to Chapter 11 of the MPS2 UFSAR and will continue to express the doses as whole body and thyroid. The licensee would not convert the doses for this system failure event to TEDE. The evaluation of the WGSF is based on the release of the maximum quantity of radioactive material allowed to be stored in the system as governed by TSs. The proposed change to an AST does not affect the TS pertinent to these systems, therefore the NRC staff agrees that no re-analysis is necessary.

A full implementation of the AST as defined in Section 1.2.1 of RG 1.183 is proposed for MPS2. Therefore, to support the licensing and plant operation changes discussed in Section 2.0 of the LAR, the licensee analyzed the following accidents employing the AST as described in RG 1.183:

1. LOCA

2. FHA

3. Cask Tip Accident

4. SGTR Accident

5. MSLB Accident

6. CREA

The licensee performed dose calculations to estimate the TEDE at the EAB for the worst 2-hour period following the onset of the accident. The integrated doses at the outer boundary of the LPZ and the integrated dose to an MPS2 CR operator were evaluated for the duration of the accident. The results of the evaluations performed by the licensee, as well as the applicable dose acceptance criteria from RG 1.183, are shown in Table 1.

The NRC staff performed independent confirmatory dose calculations for these five DBAs using the NRC-sponsored radiological consequence computer code, "RADTRAD: Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Version 3.03, as described in

NUREG/CR-6604. The RADTRAD code, developed by the Sandia National Laboratories for the NRC, estimates transport and removal of radionuclides and radiological consequence doses at selected receptors. DNC performed the radiological consequence calculations for the AST utilizing their contractor's version of the RADTRAD computer code, RADTRAD-NAI (Numerical Applications International).

3.1.1 LOCA

The design-basis LOCA radiological consequence analysis is a deterministic evaluation based on the assumption of a major rupture of the primary reactor coolant system (RCS) piping. The accident scenario assumes the failure of the emergency core cooling system (ECCS) to provide adequate core cooling which results in a significant amount of core damage, as specified in RG 1.183. This general scenario does not represent any specific accident sequence, but is representative of a class of severe damage incidents that were evaluated in the development of the RG 1.183 source term characteristics. Such a scenario would be expected to require multiple failures of systems and equipment and lies beyond the severity of incidents evaluated for design-basis transient analyses.

In the evaluation of the LOCA design-basis radiological consequence analysis, the licensee included dose contributions from the following sources:

1. Containment leakage plume
2. ECCS component leakage
3. Leakage from the refueling water storage tank (RWST) vent
4. Shine from containment and the plume
5. Shine from CR filter loading
6. RWST direct shine

During a design-basis LOCA, it is assumed that the initial fission product release to the containment will last 30 seconds and will consist of the radioactive materials dissolved or suspended in the RCS liquid. After 30 seconds, fuel damage is assumed to begin and is characterized by clad damage that releases the fission products in the fuel gap. The fuel gap release phase is assumed to continue until 30 minutes after the initial breach of the RCS. As core damage continues, the gap release phase ends and the early in-vessel release phase begins. The early in-vessel release phase continues for the next 1.3 hours. The licensee used the LOCA source term release fractions, timing characteristics, and radionuclide grouping as specified in RG 1.183 for evaluation of the AST.

The licensee generated the core radionuclide inventory for use in determining source term releases using the ORIGEN-S code. The licensee developed a core inventory based on the ORIGEN-S calculation consisting of 62 isotopes at end of fuel cycle curie levels, which formed the input for the RADTRAD-NAI dose evaluation code. The licensee used committed effective dose equivalent (CEDE) and effective dose equivalent (EDE) dose conversion factors (DCFs) from Federal Guidance Reports (FGR) 11 and 12, which is appropriate to calculate TEDE for the AST evaluation. Further, the NRC staff found that the licensee used ORIGEN-S and DCFs from FGR 11 and FGR 12 is in accordance with RG 1.183 guidance, which is acceptable to the NRC staff.

3.1.1.1 Containment Sprays

The DNC design-basis LOCA analysis credits the use of containment sprays to remove elemental and particulate iodine from the containment atmosphere. The current licensing basis credits the use of containment sprays for elemental and particulate iodine removal by the containment spray system, assuming a spray coverage area of 75.08 percent with a spray start time of 101 seconds.

The proposed AST analysis assumes that the percentage of containment that is covered by quench spray is 35.4 percent which is due to a more conservative evaluation method. Additionally, the proposed AST analysis assumes that the containment spray becomes effective at 75 seconds post-LOCA, due to an updated containment analysis. The mixing rate during spray operation is assumed to be two turnovers of the unsprayed volume per hour, which is consistent with the guidance of RG 1.183.

The licensee calculated that spray removal credit ends after 3.03 hours for elemental iodine and at 3.23 hours for particulate iodine. Further potential iodine removal due to sprays was not considered by the licensee even though the spray system may remain operating. A maximum elemental iodine decontamination factor (DF) of 199 and a maximum particulate DF of 49.5 were calculated during the period that sprays are assumed operating. The NRC staff finds that these are conservative assumptions in accordance with SRP 6.5.2 and RG 1.183.

3.1.1.2 Natural Deposition in Unsprayed Region of Containment

The licensee credited a reduction in airborne particulate radioactivity in the unsprayed region of the containment by natural deposition. The licensee used the Powers model, described in NUREG/CR-6189, “A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments,” as incorporated into the RADTRAD-NAI computer code. The licensee conservatively modeled aerosol deposition using the Powers model set at the 10^{th} percentile for the unsprayed region of the containment. This approach is acceptable to the staff because it models a lower level of removal by using the 10^{th} percentile values and, by limiting credit for natural removal to the unsprayed regions only, does not overestimate iodine removal in the containment. The Powers natural deposition model is noted in RG 1.183 as an acceptable model for use in DBA radiological consequence analysis.

3.1.1.3 Containment Leakage

The containment leakage modeled in the licensee’s radiological consequences analysis is based on the TS allowable total containment leakage (L_a) of 0.5 weight percent per day. A portion of the containment leakage is filtered by the enclosure building filtration system (EBFS), while the remainder that bypasses the EBFS is released unfiltered at ground level directly to the environment from containment. The entire containment leakage bypasses the secondary containment until the time that the EBFS drawdown is effective at 110 seconds post-LOCA. After EBFS drawdown, the bypass leak rate is assumed to be reduced to 0.007 weight percent per day. The bypass leak rate is reduced by 50 percent after 24 hours to 0.0035 weight percent per day. The filtered leak rate is 0.493 weight percent per day after 110 seconds and 0.2465 weight percent per day after 24 hours. The licensee has proposed a TS change to containment bypass leak rate to allow operational flexibility. The proposed containment bypass leak rate is double the current licensing basis leak rate value of 0.0035 weight percent per day. All other assumptions are consistent with the current licensing basis values for MPS2 and are in accordance with RG 1.183.

3.1.1.4 ECCS Leakage

During a LOCA, a portion of the fission products released from the fuel will be carried to the containment sump via spillage from the RCS, by transport of activity from the containment atmosphere to the sump by containment sprays and by natural processes such as deposition and plateout. During the initial phases of a LOCA, safety injection and the containment spray systems draw water from the RWST. Several minutes after accident initiation, valve realignment occurs to switch the suction water source for the ECCS from the RWST to the containment sump in order to recirculate the water in containment. This recirculation flow causes contaminated water to be circulated through piping and components outside of the containment where small amounts of system leakage could provide a path for the release of fission products to the environment.

To evaluate the radiological consequences of ECCS leakage, the licensee used the deterministic approach as prescribed in RG 1.183. This approach assumes that, except for the noble gases, all of the fission products released from the fuel mix instantaneously and homogeneously in the containment sump water. Except iodine, all of the radioactive materials in the sump are assumed to be in particulate form and retained in the liquid phase. As a result, the licensee assumed that the fission product inventory available for release from ECCS leakage consists of 40 percent of the core inventory of iodine. This amount is the combination of 5 percent released to the sump water during the gap release phase and 35 percent released to the sump water during the early in-vessel release phase. This source term assumption is conservative in that 100 percent of the radioiodines released from the fuel are assumed to reside in both the containment atmosphere and in the containment sump.

ECCS leakage develops when engineered safety feature (ESF) systems circulate sump water outside containment and leaks develop through packing glands, pump shaft seals and flanged connections. The licensee controls the quantity of ECCS leakage through the TS 6.13, "Systems Integrity," program limit, which calculates the maximum allowable leakage as 12 gallons per hour (gph). RG 1.183 states that the magnitude of the ECCS leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems, above which the TS, or licensee commitments, would require declaring such systems inoperable. Accordingly, the licensee used a value of 24 gph for the evaluation of the ECCS leakage contribution to the LOCA dose.

The licensee assumed that the leakage of recirculating sump fluids commences at 27.5 minutes, which is the earliest time that the recirculation of contaminated fluids would begin. Further, the licensee assumed a flashing fraction of 0.1, since the containment sump water temperature will not exceed 212 °F, consistent with the guidance of RG 1.183. In accordance with RG 1.183, the licensee assumed that the chemical form of the released iodine is 97 percent elemental and 3 percent organic.

3.1.1.5 Releases from the RWST due to ECCS Back Leakage

Following a design-basis LOCA, valve realignment occurs to switch the suction water source for the ECCS from the RWST to the containment sump. In this configuration, motor operated valves (MOVs) and check valves in the normal suction line from the RWST and MOVs in the recirculation line provide isolation between this contaminated recirculation flow stream and the RWST. Although the RWST is isolated during recirculation, design leakage through ECCS valves provides a pathway for back leakage of the containment sump water to the RWST. The RWST is located in the plant yard and is vented to the atmosphere. The licensee modeled leakage of ECCS fluid through these valves back into the RWST, with the subsequent release of the evolved iodine to the environment through the vent at the top of the RWST.

The licensee followed the guidance of RG 1.183 by assuming that the source term for release through this pathway consists only of iodine, with 97 percent assumed to be in the elemental form and the remaining 3 percent in organic form. The licensee provided a detailed evaluation of the potential leakage pathways from recirculating fluid systems back to the RWST. The licensee used the methodology approved in Millstone Power Station, Unit No. 3 (MPS3) Amendment 176 (ADAMS Accession No. ML993220168) to calculate times, flow rates, and volumes for each identified pathway. Using this methodology, the licensee determined the time for contaminated sump water to reach the RWST based on the calculated flow rates and the volume of clean water in the associated piping prior to the initiation of the recirculation phase of ECCS operation. The time required to displace the volume of clean water was reduced by 50 percent to account for mixing in the lines. The licensee considered this to be a reasonable assumption and stated that since the sump fluid is relatively cool, thermal mixing will be minimal. In addition, the licensee stated that the assumption also considered that the lines are isolated and stagnant except for minor leakage and that the mixing due to flow is negligible. The NRC staff finds these assumptions to be conservative and therefore acceptable.

The expected RWST volumetric changes were determined by the licensee based on conservative estimates of the rise in air temperature within the RWST due to solar heating. The licensee calculated an RWST vent airflow rate of 3.5 cubic feet per minute (cfm) using the ideal gas law. Since the containment sump water will not exceed 212 °F, the licensee assumed that 10 percent of the iodine in the sump water that leaks back to the RWST will be released to the environment at the RWST airflow rate, in accordance with RG 1.183. Additionally, the licensee calculated the time for back leakage to reach the RWST at 6.45 hours post-LOCA.

3.1.1.6 LOCA Analysis Results

The licensee added the dose results from each of the above pathways to determine the total LOCA radiological consequences. The licensee's modeling of the CR and the associated analysis inputs and assumptions are discussed below in Section 3.2. The licensee evaluated the radiological consequences resulting from the postulated LOCA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose requirements provided in 10 CFR 50.67 and accident dose criteria specified in SRP 15.0.1. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the staff are presented in Table 5 and the licensee's calculated dose results are given in Table 1. The NRC staff performed independent confirmatory dose evaluations to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the LOCA meet the applicable accident dose criteria and are, therefore, acceptable.

3.1.2 FHA

The FHA analysis postulates that a spent fuel assembly is dropped during fuel handling. All of the fuel rods in the dropped assembly are conservatively assumed to experience fuel cladding damage, releasing the radionuclides within the fuel rod gap to the fuel pool or reactor cavity water. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released because of the accident. Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the reactor cavity or spent fuel pool (SFP), depending on their physical and chemical form. DNC assumed no decontamination for noble gases, an effective DF of 200 for radioiodines, and retention of all aerosol and particulate fission products in the overlying water. As prescribed in RG 1.183, the FHA is analyzed based on the assumption that 100 percent of the fission products released from the reactor cavity or SFP are released to the environment in 2 hours. The licensee did not credit filtration, holdup, or dilution of the released activity. Since the revised assumptions and inputs are identical for the FHA within containment and the FHA outside containment, the results of the two events are identical. The assumptions pertaining to the source term for the FHA have not been changed from the FHA assumptions that were recently approved in Amendment 284 (ADAMS Accession No. ML042360671) for selective implementation of the AST.

In the revised FHA, the licensee made changes to the following elements of the accident analysis that pertain to the CR habitability analysis:

1. CR atmospheric dispersion (χ/Q) factors
2. CR inleakage assumptions
3. CR filtration efficiency
4. CR automatic isolation timing
5. CR emergency ventilation (CREV) system manual initiation timing

The evaluation of the DNC χ/Q factors are discussed in Section 3.3.

For the FHA, the licensee assumes that it will take 20 seconds for the CR to isolate following detection of released activity by the MPS2 CR inlet radiation monitors. The licensee proposes a new TS limit for CR envelope unfiltered inleakage of 200 cubic feet per minute (cfm), which bounds the measured inleakage of < 130 cfm, as determined by tracer gas testing. The current TS value is 130 cfm.

For the FHA, the licensee assumes that the CR will experience 20 seconds of normal ventilation intake flow at 800 cfm prior to CR isolation. After CR isolation, the licensee assumes unfiltered inleakage at 200 cfm. The licensee assumes a 1 hour period at a neutral pressure condition that accounts for the operator action to align the safety-related CREV system. The licensee assumes that the CREV system is actuated manually at 1 hour, providing a CR recirculation flow rate of 2250 cfm. The filter efficiencies for the CREV system assumed by the licensee are 90 percent for elemental and aerosol iodine species, and 70 percent for organic iodines. The design-basis filter efficiency for the CREV system is 90 percent for all iodine species, therefore the licensee’s filter efficiency assumption is conservative.

The licensee evaluated the radiological consequences resulting from the postulated FHA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose requirements provided in 10 CFR 50.67 and accident dose criteria specified in SRP 15.0.1. The staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance, identified in Section 2.0 of this SE. The assumptions found acceptable to the staff are presented in Table 6 and the licensee's calculated dose results are given in Table 1. The NRC staff performed independent confirmatory dose evaluations to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the FHA meet the applicable accident dose criteria and are, therefore, acceptable.

3.1.3 Cask Tip Accident

The cask tip accident analysis postulates that a cask tips over in the SFP and damages impacted fuel assemblies. Damage occurs to 1560 fuel assemblies. Of these, 184 have a decay time of 1 year and the remainder have decayed for 5 years. Administrative controls limit the age of fuel assemblies in the area of potential impact. The damaged assemblies are conservatively assumed to experience fuel cladding damage, releasing the radionuclides within the fuel rod gap to the SFP. The fission product inventory in the fuel rod gap of the damaged assemblies is assumed to be instantaneously released because of the accident. Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the SFP, depending on their physical and chemical form. DNC assumed no decontamination for noble gases, an effective DF of 200 for radioiodines, and retention of all aerosol and particulate fission products in the overlying water. The cask tip accident, similar to the FHA, is analyzed based on the assumption that 100 percent of the fission products released from the SFP are released to the environment in 2 hours. The licensee did not credit filtration, holdup, or dilution of the released activity. The assumptions pertaining to the source term for the cask tip accident have not been changed from the cask tip accident assumptions that were recently approved in Amendment 284 for selective implementation of the AST.

In the revised cask tip accident, the licensee made changes to the following elements of the accident analysis that pertain to the CR habitability analysis:

1. CR χ/Q factors
2. CR inleakage assumptions
3. CR filtration efficiency
4. CR automatic isolation timing
5. CREV system manual initiation timing

The evaluation of the DNC χ/Q factors are discussed in Section 3.3.

For the cask tip accident, the licensee assumes that it will take 20 seconds for the CR to isolate following detection of released activity by the MPS2 CR inlet radiation monitors. The licensee assumes that the CR will experience 20 seconds of normal ventilation intake flow at 800 cfm prior to CR isolation. After CR isolation, the licensee assumes unfiltered inleakage at 200 cfm. The licensee assumes a 1 hour period at a neutral pressure condition that accounts for the operator action to align the safety-related CREV system. The licensee assumes that the CREV system is actuated manually at 1 hour, providing a CR recirculation flow rate of 2250 cfm. The filter efficiencies for the CREV system assumed by the licensee are 90 percent for elemental and aerosol iodine species, and 70 percent for organic iodines. The design-basis filter efficiency for the CREV system is 90 percent for all iodine species, therefore the licensee's filter efficiency assumption is conservative.

The licensee evaluated the radiological consequences resulting from the postulated cask tip accident and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose requirements provided in 10 CFR 50.67 and accident dose criteria specified in SRP 15.0.1. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance, identified in Section 2.0 of this SE. These assumptions are presented in Table 6 and the licensee's calculated dose results are given in Table 1. The NRC staff performed independent confirmatory dose evaluations to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the cask tip accident meet the applicable accident dose criteria and are, therefore, acceptable.

3.1.4 SGTR Accident

The licensee evaluated the radiological consequences of an SGTR accident as a part of the full implementation of the AST. In an SGTR accident, it is assumed that there is a complete severance of a single steam generator (SG) tube. The accident is assumed to take place at full power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited amount of fuel damage. The postulated break allows primary coolant liquid to leak to the secondary side of the ruptured SG (denoted by the licensee as the affected SG) with an assumed release to the environment through the SG atmospheric dump valves (ADVs) or main steam safety valves (MSSVs). For this accident scenario, a loss of offsite power (LOOP) is assumed to occur concurrently with the tube rupture. Because the LOOP renders the main condenser unavailable, the plant is cooled down by release of steam to the environment. In the DNC analysis, the ADV/MSSV on the affected SG is assumed to open to control SG pressure at the beginning of the event. The affected SG discharges steam to the environment for 1 hour until the generator is manually isolated.

The licensee referred to the unaffected SG as the intact SG. The licensee evaluated the dose consequences from discharges of steam from the intact SG for a period of 17 hours, until the primary system has cooled sufficiently to allow placing the shutdown cooling system (SDC) in service. After a period of 17 hours, SDC is capable of removing 100 percent of the decay heat. At this point in the accident sequence, steaming is no longer required for cool down and release from the intact SG is terminated.

Appendix F of RG 1.183 identifies acceptable radiological analysis assumptions for an SGTR accident. If a licensee demonstrates that no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by TS. Two radioiodine spiking cases are considered. The first case is referred to as a pre-accident iodine spike and assumes that a reactor transient has occurred prior to the postulated SGTR that has raised the primary coolant iodine concentration to the maximum value permitted by the TS for a spiking condition. For MPS2, the maximum iodine concentration allowed by TS as a result of an iodine spike is 60 microcuries per gram (μCi/gm) dose equivalent (DE) I-131.

The second case assumes that the primary system transient associated with the SGTR causes an iodine spike in the primary system. This case is referred to as a concurrent iodine spike.

The increase in primary coolant iodine concentration for the concurrent iodine spike case is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the TS limit for normal operation. For MPS2, the RCS TS limit for normal operation is 1 μCi/gm DE I-131.

The licensee's evaluation indicates that no fuel damage is predicted as a result of an SGTR accident. Therefore, consistent with the current licensing basis analysis and regulatory guidance, the licensee performed the SGTR accident analyses for the pre-accident iodine spike case and the concurrent accident iodine spike case. The licensee also modeled the gross gamma activity in the primary coolant at the TS limit of $100/E_{bar}$. Modeling of the CR is discussed below in Section 3.2.

3.1.4.1 Releases from the Affected SG

The licensee assumed that the source term resulting from the radionuclides in the primary system coolant, including the contribution from iodine spiking, is transported to the affected SG by the break flow. The licensee's analysis determined that the break flow is terminated after 1 hour. A portion of the break flow is assumed to flash to steam because of the higher enthalpy in the RCS. The noble gas and iodine in the flashed portion of the break flow will ascend to the steam space of the affected generator and be available for release, with no credit taken for scrubbing by the SG liquid. The radionuclides entering the steam space as the result of flashing pass directly to the environment through the SG ADVs/MSSVs. The iodine and other non-noble gas isotopes in the non-flashed portion of the break flow are assumed to mix uniformly with the SG liquid mass and then released to the environment in direct proportion to the steaming rate and in inverse proportion to the applicable partition coefficient (PC) that models retention in the SG liquid. In accordance with the guidance from RG 1.183, the licensee's evaluation of the releases from the steaming of the liquid mass in the SG credits a PC of 100 for iodines and 250 for all other non-noble gas isotopes. Thus, 1 percent of the elemental and organic iodines and 0.4 percent of the particulates are released from the SG liquid to the environment along with the steam flow. Following the applicable regulatory guidance, the licensee assumed that all noble gas radionuclides released from the primary system are released to the environment without reduction or mitigation.

3.1.4.2 Releases from the Intact SG

The licensee assumed that the source term resulting from the radionuclides in the primary system coolant, including the contribution from iodine spiking, is transported to the intact SG by the leak rate limiting condition for operation (LCO) of 1 gpm specified in the TS. All radionuclides in the primary coolant leaking into the intact SG are assumed to enter the SG liquid. Radionuclides initially in the SG liquid, and those entering the SG liquid from the leakage flow, are released as a result of secondary liquid boiling/steaming, with a PC of 100 for iodines and a PC of 250 for all non-noble gas isotopes. Therefore, 1 percent of the iodines and 0.4 percent of the particulates are assumed to pass into the steam space and then directly to the environment. The licensee assumed that all noble gases that are released from the primary system to the intact SG are released to the environment without reduction or mitigation. Releases were assumed to continue from the intact SG for a period of 17 hours until the SDC is able to remove 100 percent of decay heat with no requirement for steaming to augment cooldown.

3.1.4.3 SGTR Analysis Results

The licensee evaluated the radiological consequences resulting from both cases for the postulated SGTR accident and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose requirements provided in 10 CFR 50.67 and accident dose criteria specified in SRP 15.0.1. As discussed above, the NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. These assumptions are presented in Table 7 and the licensee's calculated dose results are given in Table 1. Additionally, the NRC staff performed independent confirmatory dose evaluations to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the SGTR accident meet the applicable accident dose criteria and are, therefore, acceptable.

3.1.5 MSLB Accident

The licensee evaluated the radiological consequences of an MSLB accident as a part of the full implementation of the AST. The MSLB accident considered is the complete severance of a main steam line. The licensee considered the radiological consequences of a MSLB break in the following structures: 1) turbine building, 2) containment, and 3) enclosure building. The licensee also separates the accidents into two categories: 1) in containment, 2) outside containment. Outside containment describes an MSLB located in the turbine building or enclosure building structures.

The licensee's evaluation indicates that no fuel damage is predicted as a result of an MSLB accident outside containment. Therefore, consistent with the current licensing analysis basis and RG 1.183, the licensee performed the MSLB accident outside containment analyses assuming that the accident occurs at full power with both the primary and secondary coolant concentrations at their TS limit for operation. As in the SGTR accident, the licensee's MSLB outside containment evaluation includes the effects of primary system iodine spiking for both the pre-accident iodine spike case and the concurrent iodine spike case. The spiking cases are as described in Section 3.1.4 for the SGTR with the following exception. For the MSLB accident outside containment, the concurrent iodine spike is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the TS limit for normal operation. In effect, it is assumed that the MSLB accident outside containment results in a more severe concurrent iodine spike than the SGTR accident. The duration of the concurrent iodine spike is assumed to be 8 hours, in accordance with the applicable guidance. The licensee also modeled the gross gamma activity in the primary coolant at the TS limit of $100/E_{bar}$.

For a MSLB in containment, a 3.7 percent fuel failure is assumed. This source term is used in conjunction with a 1.69 fuel peaking factor. The licensee proposed this peaking factor, which is slightly higher (more conservative) than the current licensing basis peaking factor of 1.65, for added margin. The source term assumption is the same as in the MPS2 current licensing basis, and is acceptable.

The MSLB accident begins with a break in one of the main steam lines leading from a SG to the turbine. The SG that experiences a secondary side depressurization as a result of an MSLB is referred to as being in a faulted condition. The licensee uses the term "affected SG" to describe the faulted SG. The affected SG is assumed to release steam for 36 hours, which is the time required for the RCS to be cooled down to 212 °F. The 36-hour steaming period is based on the time necessary to cooldown to 212 °F by crediting safety grade equipment only. For both the MSLB scenarios, the licensee assumed that dry out of the affected SG will occur 750 seconds after the MSLB.

The licensee evaluated each scenario of the MSLB accident assuming a concurrent LOOP. Due to the assumption of a LOOP, the condenser is unavailable and cool down of the primary system is accomplished through the release of steam from the intact SG ADVs or MSSVs. The release from the intact SG continues for 16 hours through the ADVs/MSSVs until SDC can fully remove decay heat. Modeling of the CR is discussed below in Section 3.2.

The licensee proposed a new maximum amount of primary-to-secondary leakage allowed by the TS to any one SG of 75 gallons per day (gpd). The licensee conservatively assigned leakage of 150 gpd between the intact and affected SG to maximize the calculated dose consequence.

3.1.5.1 Release from Affected SG (Break Outside Containment)

For the affected SG, DNC assumed the release passes directly into the turbine building or enclosure building with no credit taken for holdup, partitioning or scrubbing by the SG liquid. The licensee did not take credit for any holdup or dilution in the turbine building or enclosure building. The DNC analysis assumes the release into the turbine building or enclosure building is exhausted to the environment and subsequently transported from the environment into the CR assuming conservative χ/Q factors.

The licensee assumed that during the first 750 seconds of the accident, the affected SG steams dry as a result of the MSLB in the turbine or enclosure building, releasing all of the nuclides in the secondary coolant that were initially contained in the SG. The licensee conservatively assumed that during the first 36 hours, primary coolant leaks into the affected SG at the rate of 150 gpd directly releasing all of the coolant activity to the environment. For the turbine building location, this release is assumed to continue for 36 hours, until the RCS has cooled to below 212 °F, at which time the release from this pathway terminates. For the enclosure building location, this release is assumed to continue for 720 hours, until the accident is over. The licensee used a transport model for noble gases, iodine and particulates that is consistent with the guidance in Appendix E of RG 1.183.

3.1.5.2 Release from Affected SG (Break Inside Containment)

For the MSLB in containment scenario, the licensee assumed during the first 750 seconds of the accident, the affected SG steams dry as a result of the MSLB, releasing all of the nuclides in the secondary coolant that were initially contained in the SG. The licensee conservatively assumed that during the first 36 hours, primary coolant leaks into the affected SG at the rate of 150 gpd directly releasing all of the coolant activity to the containment. Initially all of the releases from the containment bypass the secondary containment at the TS limit of 0.5 percent per day. At 250 seconds the EBFS draws down the enclosure building and becomes effective at collecting and filtering the release. Once EBFS becomes effective, the licensee split the release into filtered and bypass portions. The licensee released the bypass portion at ground level unfiltered at the proposed TS limit of 0.007 percent per day. The licensee treated the filtered portion as an elevated release at a rate of 0.493 percent per day. After 24 hours, both leak rates are reduced by 50 percent, which is consistent with RG 1.183.

3.1.5.3 Release from Intact SG

In order to ensure proper accounting of gross gamma, iodine and noble gas releases from the intact SG, the licensee evaluated all the significant nuclide transport models for the MSLB accident. The licensee evaluated the release of the activity from the primary coolant leaking into the intact SG volume at a primary-to-secondary leak rate of 150 gpd. Radionuclides initially in the SG liquid and those entering the SG from the primary-to-secondary leakage flow are released as a result of secondary liquid boiling/steaming. An assumed Partition Coefficient of 100 results in 1 percent of the elemental and organic iodines in the SG bulk liquid being released to the environment at the steaming rate. An assumed moisture carryover of 0.4 percent results in a Partition Coefficient of 250 for particulates in the SG bulk liquid released to the environment. Radionuclides initially in the steam space do not provide any significant dose contribution. The transport to the environment of noble gases from the primary coolant is assumed to occur without any mitigation or holdup.

3.1.5.4 MSLB Analysis Results

The licensee evaluated the radiological consequences resulting from the postulated MSLB accident occurring in the containment in the enclosure building or in the turbine building, and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose requirements provided in 10 CFR 50.67 and accident dose criteria specified in SRP 15.0.1. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. These assumptions are presented in Table 8 and the licensee's calculated dose results are given in Table 1. The NRC staff performed independent confirmatory dose evaluations to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the MSLB meet the applicable accident dose criteria and are, therefore, acceptable. Additionally, the NRC staff finds that the licensee has proposed TS limits on primary-to-secondary leakage commensurate with the revised analysis.

3.1.6 CREA

The CREA 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. For this accident, localized damage to fuel cladding is projected. The licensee assumed that as a result of localized fuel cladding damage, 10 percent of the gap activity from the damaged fuel is released to the primary coolant, consistent with the MPS2 current licensing basis. The mechanical failure breeches the reactor pressure vessel head resulting in a release of primary coolant to the containment atmosphere. Releases to the environment are assumed to occur through two separate pathways:

- Release of containment atmosphere (methodology and assumptions similar to the LOCA).

- Release of RCS inventory via primary-to-secondary leakage through SGs.

To evaluate the release to containment atmosphere, the licensee employed the guidance from Appendix H of RG 1.183. DNC assumed that 11.5 percent of the fuel rods fail, releasing the fission product inventory in the fuel rod gap. The licensee assumed that 10 percent of the core inventory of iodines and noble gases is in the fuel rod gap. Therefore, for the fuel clad failure, the fraction of core activity released is 0.0115 for both halogens and noble gases.

DNC has determined that containment sprays will not initiate due to a CREA and, as a result, the licensee did not evaluate dose contributions from ECCS leakage and RWST back leakage as in the LOCA analysis. For the release into containment resulting from the CREA the licensee did not credit natural deposition as was done in the LOCA analysis. The licensee assumed that the containment leak rate is reduced by 50 percent at 24 hours for both the offsite and the CR analyses. The licensee has determined that a safety injection signal will be initiated 1 minute after the accident, followed by a 25 second delay due to diesel startup for the EBFS actuation signal to initiate enclosure building isolation and then an additional 20 seconds for the CR to isolate. As a result, the CR will not be isolated until 105 seconds following a CREA.

The second release path evaluated by the licensee is via the secondary system. The licensee based the evaluation of the activity in the secondary system release on the guidance in Appendix H of RG 1.183. The licensee assumed primary to secondary leak rate of 150 gpd exists until shutdown cooling is in operation and releases from the SGs terminate at approximately 16 hours. A LOOP is conservatively assumed to occur concurrent with the initiation of the CREA, rendering the main condenser unavailable. With the main condenser unavailable, the plant is cooled down by releases of steam to the environment via the ADVs and/or the MSSVs. The licensee assumed the chemical form of the iodines released from the SGs to be 97 percent elemental and 3 percent organic as is consistent with the applicable guidance. As in the evaluation of the MSLB accident, the licensee assumed an iodine partition factor of 100 in the SGs and assumed that the noble gas activity released to the secondary system is released to the environment without reduction or mitigation. Modeling of the CR is discussed below in Section 3.2.

The licensee evaluated the radiological consequences resulting from the postulated CREA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose requirements provided in 10 CFR 50.67 and accident dose criteria specified in SRP 15.0.1. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. These assumptions are presented in Table 9 and the licensee's calculated dose results are given in Table 1. The NRC staff performed independent confirmatory dose evaluations to ensure a thorough understanding of the licensee's methods. The EAB, LPZ, and CR doses estimated by the licensee for the CREA were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.2 CR Habitability

The CR effective volume used in the habitability analyses is 35,656 ft^3. The normal CR ventilation intake flow rate prior to isolation is 800 cfm. The licensee has proposed a new manual operator action to isolate the CR after 4 hours for the MSLB accident. The CR will isolate on a safely injection actuation signal for the LOCA and CREA. For all other DBAs analyzed in this LAR, the CR will isolate on the CR ventilation radiation monitor.

The period after CR isolation and prior to CREV system initiation is referred to as the CR neutral condition. During the neutral condition, the CR is isolated, the normal ventilation intake flow rate of 800 cfm has terminated, and the CREV system is not operating. During the neutral condition there is no mechanically induced ventilation of the CR. For the neutral condition, the licensee has proposed an increase in the TS unfiltered inleakage to 200 cfm. The licensee assumes a 1 hour delay after CR isolation for manual alignment of the CREV system to place into filtered recirculation. The 200 cfm of unfiltered inleakage is assumed to continue while the CREV system is in operation.

The CREV system provides 2250 cfm of filtered recirculation flow to the CR. Once initiated, CREV system operates for the duration of the event in the CR habitability analyses. The CREV system filter iodine removal efficiencies, as assumed by the licensee, are 90 percent aerosol, 90 percent elemental, and 70 percent organic. The licensee assumed CR occupancy factors of 100 percent for the first 24 hours, 60 percent from 24 to 96 hours, and 40 percent from 96 to 720 hours.

The licensee evaluated the CR habitability for a period of 720 hours notwithstanding the relatively short duration of several of the DBAs evaluated; most notably the FHA, which is assumed to end in 2 hours.

The licensee used the current licensing basis methodology and sources (CR filter shine, cloud shine, RWST direct shine, and containment direct shine) to calculate the contribution to CR dose from external sources, which is acceptable to the NRC staff.

3.3 Atmospheric Dispersion Estimates

DNC used onsite meteorological data collected during calendar years 1997-2001 to generate new CR χ/Q values for the MPS2 ground level releases associated with the dose assessments discussed above. The resulting χ/Q values represent a change from the χ/Q values used in the current MPS2 Final Safety Analysis Report, Chapter 14, "Safety Analysis." The licensee used previously generated χ/Q values for postulated elevated releases from the Millstone stack to the MPS2 CR as well as all releases to the EAB and LPZ.

3.3.1 Meteorological Data

As discussed in section 3.3, DNC generated new CR χ/Q values for the MPS2 ground level release dose assessments using onsite meteorological data collected from 1997-2001. These data were provided for staff review in the form of hourly meteorological data files formatted for input into the ARCON96 χ/Q computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes"). The NRC Staff performed a quality review of the 1997-2001 hourly meteorological databases using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data." Further review was performed using computer spreadsheets.

Wind speed and wind direction data were measured on the Millstone onsite meteorological tower at heights of 10.1 meters and 43.3 meters above the ground. Temperature difference data, which were used to determine atmospheric stability class, were measured between the 43.3-meter and 10.1-meter levels. The combined data recovery of the wind speed, wind direction, and stability data was in the upper 90 percentiles at both levels during each of the

5 years. This meets the data recovery recommendation of RG 1.23, “Onsite Meteorological Programs.”

The NRC staff has judged that the reported Millstone 1997-2001 atmospheric stability measurements result in a conservative estimate of the χ/Q values when applied to the dose assessment for this LAR because of the number of reported stable atmospheric conditions. Wind speed and wind direction frequency distributions for each measurement channel were similar from year to year and when comparing measurements at the 10.1-meter and 43.3-meter levels.

For the reasons cited above, the NRC staff has concluded that the 1997-2001 meteorological data measured at the Millstone site provide an acceptable basis for making χ/Q estimates for use in the dose assessments performed in support of this specific LAR. However, these data should not be considered acceptable for use in other dose assessments or other meteorological applications without further review.

3.3.2 Control Room Atmospheric Dispersion Factors

MPS2 has a single CR air intake and postulated releases to the CR were modeled from the closest point on the containment enclosure building, RWST, closest ADV and enclosure building blowout panels, closest turbine building blow out panel, closest group of MSSVs and the Millstone stack. Releases from the stack were modeled as an elevated release. Releases from the containment enclosure building were modeled as a ground level diffuse release. All other releases were assumed to be ground level point source releases.

DNC generated new CR χ/Q values for postulated ground level releases using guidance provided in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants." These new CR χ/Q values were calculated using the ARCON96 computer code. RG 1.194 states that ARCON96 is an acceptable methodology for assessing CR χ/Q values for use in design-basis accident radiological analyses. the NRC staff evaluated the applicability of the ARCON96 model and concluded that there are no unusual siting, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude use of this model in support of this LAR for MPS2.

In its March 6, 2007, response to NRC staff questions, DNC provided an analysis of each design-basis accident to demonstrate that it had addressed limiting cases for each source/receptor pair including scenarios involving LOOP and other single failures. In addition, when modeling the ground level releases, the licensee used a single straight line horizontal distance input in all cases except for postulated releases from the RWST. Because of the intervening location of the enclosure building with respect to the path between the RWST and the CR air intake, DNC used the sum of two horizontal straight line distances as the input for the RWST calculation.

The licensee used previously generated control room χ/Q values for elevated releases from the Millstone stack as addressed in the SE associated with MPS2 License Amendment No. 228 dated March 10, 1999 (ADAMS Accession No. ML9903250121). However, NRC staff notes two apparent differences between the values associated with Amendment 228 and those used by the licensee in support of the current LAR. First, the 0-4 hour χ/Q value which appears in Table

1 of the NRC staff's SE contains a typographical error which has been corrected by letter dated April 13, 2007 (ADAMS Accession No. ML070890279).

Second, the 24-96 hour and 96-720 hour χ/Q values supporting Amendment 228 appear to be less than the values used in the dose assessment for this LAR. The licensee explained in its March 6, 2007 response to NRC staff questions that the χ/Q values supporting Amendment 228 incorporate a control room occupancy factor of 60 percent for the first time period (24-96 hours) and 40 percent for the second time period (96-720 hours) which is consistent with NRC guidance. The χ/Q values for these two time periods were reported in the licensee's submittal for the current LAR with the occupancy factors removed to be consistent with the new χ/Q values that were calculated using ARCON96. Control room occupancy factors were then applied as a subsequent step in the dose assessment. Thus, the control room χ/Q values for postulated releases from the Millstone stack are those approved as part of Amendment 228.

In summary, the NRC staff qualitatively reviewed the inputs to the ARCON96 calculations and found them generally consistent with site configuration drawings and staff practice. The staff made an independent evaluation of the resulting χ/Q estimates by running the ARCON96 computer model for several random cases and obtained similar results. On the basis of this review, the NRC staff has concluded that the χ/Q values for the MPS2 LOCA, FHA, SGTR, MSLB, and CREA releases to the CR air intake presented in Table 2 are acceptable for use in the design-basis accident assessment performed in support of this LAR.

3.3.3 Offsite Atmospheric Dispersion Factors

DNC used design-basis χ/Q values that were accepted by the NRC staff in MPS2 License Amendment 228 to evaluate the impact of the MPS2 postulated ground level radiological releases to the EAB and LPZ. Elevated release χ/Q values were those previously addressed in the SE associated with MPS3 License Amendment No. 211 dated September 16, 2002 (ADAMS Accession No. ML022470399). Use of the elevated release χ/Q values from the MPS3 Amendment 211 analysis is acceptable because MPS2 and MPS3 share the single Millstone site stack. Thus, the χ/Q of effluent releases from the stack to the EAB and LPZ is the same whether the effluent is generated by MPS2 or MPS3. The EAB and LPZ χ/Q values are provided in Table 3.

3.4 TS Changes

3.4.1 TS Definitions, "Dose Equivalent I-131"

Following the guidance in Section 4.1.2 of RG 1.183, the licensee used the DCFs listed in FGR 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," to evaluate the dose from the inhalation of radioactive material in the AST DBA radiological analyses. Previous analyses incorporated DCFs from RG 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I." To maintain consistency between the revised DBA analyses and the TS, the licensee has proposed to revise the definition of DE I-131, in Section 1.19 of the TS definitions, to reference FGR 11 as the source of thyroid DCFs, and delete reference to RG 1.109, Revision 1. The following proposed revision is acceptable to the staff because it follows the guidance contained in RG 1.183.

> DOSE EQUIVALENT I-131 shall be that concentration of I-131(microCurie/gram) which alone would produce the same CEDE-thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133,1-134, 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."

3.4.2 TS 3/4.3, "Instrumentation"

The licensee proposed to revise the radiation monitoring LCO 3.3.3.1.b from:

> 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.

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.

Prior to this amendment request, TS 3/4.3.3, Table 3.3-6 and associated Action 16 were changes for conformity with actual plant conditions via Amendment Nos. 289 and 291 (ADAMS Accession Nos. ML052700062 and ML060760208, respectively).

The proposed revision to the wording of the LCO will continue to require the minimum number of channels assumed to be OPERABLE in the MPS2 design-basis and is therefore acceptable.

3.4.3 TS 3/4.3.3, "Monitoring Instrumentation"

The licensee proposed to delete TS 3/4.3.4, "Containment Purge Valve Isolation Signal," and the associated surveillance test requirements. The licensee stated that the containment purge valve isolation signal (CPVIS) is not credited in the accident analyses described in the AST analyses. The NRC staff reviewed the request to remove current TS 3/4.3.4 against the criterion of 10 CFR 50.36(c)(2)(ii), which provides the regulatory requirements for the contents of the TSs.

In accordance with LCO 3.6.3.2, the containment purge supply and exhaust isolation valves are maintained as sealed closed containment isolation valves during MODES 1 through 4. As discussed in the SE for MPS2 Amendment 216 (ADAMS Accession No. ML012920154), the valves have no manual actuator and therefore, cannot be locked closed. To meet the requirement of being locked closed, the control power fuses are removed for a closed valve and the access to the fuse blocks is locked.

LCO 3.6.3.2 ensures that the purge supply and exhaust valves remain closed in MODES 1 through 4 regardless of whether a CPVIS exists. Further, SR 4.6.3.2 requires that the containment purge supply and exhaust isolation valves be verified sealed closed at least once per 31 days. The LCO and SR provide adequate assurance that the containment isolation valves support containment integrity during MODES 1 through 4. Additionally, the 10 CFR,

Appendix J Type C leak rate testing is periodically performed on the containment purge supply and exhaust valves to verify that the valve overall containment leakage is within assumed limits.

The NRC staff finds that the existing TS requirements in LCO 3.6.3.2 and SR 4.6.3.2, in addition to the periodic Type C leak rate testing will maintain the valves in their analytically assumed states. Therefore, the requirements of TS 3/4.3.4 do not meet the criterion of 10 CFR 50.36(c)(2)(ii) and may be removed from the TSs.

3.4.4 TS 3/4.4.6, "Reactor Coolant System Leakage"

The primary-to-secondary leakage rate in TS LCO 3.4.6.2.c is being revised from 0.035 gpm (equivalent to 50.4 gpd) to 75 gpd. The current design basis analyses limit is also 0.035 gpm (50.4 gpd). However, the licensee's AST implementation analyses assumed a higher leakage rate of 150 gpd. The use of 150 gpd in the associated AST analyses provides a conservative margin beyond the proposed TS leakage rate of 75 gpd. Since the acceptance criteria is met for the AST implementation analyses assuming the leakage rate to be 150 gpd, the NRC staff finds that this proposed change is consistent with the DBA radiological consequence analysis and is acceptable for use with respect to the radiological consequences of DBAs.

3.4.5 TS 3/4.6.1, "Primary Containment"

The licensee's AST implementation analyses assumed the leakage rate acceptance criteria for all penetrations that are secondary containment bypass leakage paths to be equal to 0.014 L_a. Therefore, to maintain consistency between the DBA radiological analyses and the TS, the licensee proposed to change the leakage rate acceptance criteria for all penetrations that are secondary containment bypass leakage paths in Section 3.6.1.2.c from < 0.0072 L_a to < 0.014 L_a. Since the acceptance criterion is met for the AST implementation analyses assuming a leakage rate of 0.014 L_a for all penetrations that are secondary containment bypass leakage, the staff finds this proposed change is consistent with the DBA radiological consequence analysis and is acceptable with respect to the radiological consequences of DBAs.

3.4.6 3/4.7.6, "Control Room Emergency Ventilation System"

The licensee proposed to revise the applicability statement 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

The proposed change removes the potential for misinterpretation that the TS is not applicable in MODES 5 and 6 if irradiated fuel is not being moved. The proposed wording will continue to hold TS 3/4.7.6 applicable in the appropriate MODES of plant operation and during appropriate plant activities, therefore, the proposed change is acceptable.

Additionally, with the implementation of AST, the licensee assumed the CR air in-leakage rate to be equal to 200 cfm with the CREV system operating in the recirculation/filtration mode. Therefore to maintain consistency between the DBA radiological analyses and the TS, the licensee proposed to change the leakage rate acceptance criteria for surveillance requirement 4.7.6.1.e.3 from <130 cfm to < 200 cfm with the CREV system operating in the recirculation/filtration mode. Since the acceptance criterion is met for the AST implementation analyses assuming a leakage rate of 200 cfm for CR in-leakage, the NRC staff finds this proposed change is consistent with the DBA radiological consequence analysis and is acceptable with respect to the radiological consequences of DBAs.

3.4.7 SR 4.7.6.1.e.3, "Control Room Emergency Ventilation System"

The licensee proposed to revise SR 4.7.6.1.e.3 such that the limit on control room air in-leakage is less than 200 SCFM vice the currently required 130 SCFM. An increase in the allowed control room in-leakage in SR4.7.6.1.e.3 will result in an increased radiological loading of the emergency ventilation system filter trains' charcoal beds. Regulatory Guide 1.52 (Section 4.10) states that the design of the adsorber section in ESF atmospheric cleanup system "… should consider possible iodine desorption and adsorbent auto-ignition that may result from radioactivity-induced heat in the adsorbent and concomitant temperature rise."

The MPS2 UFSAR addresses the issue of filter bed fire hazards in Section 9.9.10.4 which states in part:

> "The charcoal filter elements within the CRFS are analyzed to ensure adequate residual heat removal capabilities following any single failure. The analysis concludes that the maximum temperature calculated, based on a radioactive filter inventory which was conservatively assumed to be ten (10) times greater than the maximum inventory calculated resulting from a design basis accident at the site, was less than 212°F (100°C). This is substantially below the charcoal ignition temperature, thus, filter bed isolation should not constitute a fire hazard. Temperature indication is provided to alert personnel of excessive charcoal bed temperature."

Based on this, the staff concludes that the charcoal beds are of adequate design to adsorb the potential increased radiological loading resulting from a 70 SCFM increase in the allowed control room in-leakage. As noted in section 3.4.6 above, the acceptance criterion is met for the AST implementation analyses assuming a leakage rate of 200 cfm for CR in-leakage. Therefore, the NRC staff finds this proposed change is consistent with the DBA radiological consequence analysis and is acceptable with respect to the radiological consequences of DBAs.

3.5 Human Factors

As discussed previously, DNC re-analyzed selected DBAs pursuant to 10 CFR 50.67. Contained within this re-analysis were two assumptions regarding manual operator actions that differed from the current licensing basis:

1. A timing change to an existing operator action, such that within 1 hour of CR isolation, instead of within the currently credited time of 10 minutes, operators will place the CREV system in filtered recirculation.

2. A new operator action to isolate the CR within 4 hours after a MSLB outside containment.

Using the guidance contained in NRC IN 97-78, ANSI/ANS 58.8-1994, and NUREG-0800, the staff evaluated the proposed changes with respect to human performance.

For the first proposed change, the timing change to 1 hour for placing CR filtered recirculation in service, a detailed review was not conducted. This manual operator action is already credited in MPS2's UFSAR, although the current licensing basis is for operators to place the CREV system in filtered recirculation within 10 minutes of CR isolation. Since DNC's request represents a relaxation of the currently credited operator action time from 10 minutes to 1 hour, a detailed review was not necessary. The NRC has determined that this proposed relaxation in operator action time is acceptable.

The second proposed change, to isolate the CR within 4 hours after a MSLB outside containment, however, was a new operator action, and is not a part of the current licensing basis at MPS2. For this new manual operator action, a detailed review was conducted and it is described below.

3.5.1 Diagnosis of a MSLB by CR Alarms and Indications

CR indications and alarms used by plant operators to identify and diagnose a MSLB outside containment include:

- Decreasing steam generator pressure, with no radiation increase, and no increase in containment pressure
- Increasing feedwater flow
- Steam flow/feed flow mismatch
- Decreasing electrical output of the main generator, with increasing reactor power
- Decreasing RCS average temperature
- Decreasing RCS pressure

For large breaks, the following indications will also be used:

- High differential pressure in the enclosure building
- Safety injection actuation signal (SIAS) from low RCS pressure
- The SIAS will also initiate a containment isolation actuation signal (CIAS) and an enclosure building filtration actuation signal (EBFAS)

With the exception of the high differential pressure in the enclosure building, all of the above indications and alarms are safety grade and powered from safety grade electrical power supplies. Operators are routinely trained and evaluated on their ability to properly diagnose and respond to MSLBs, utilizing the above indications. Therefore, there is a high likelihood that operators will be able to identify the occurrence of a MSLB.

3.5.2 Complexity of Isolating the CR

To manually isolate the CR, one hand switch is required to be operated from a CR air conditioning system (CRACS) panel located in the main CR. Operating this one hand switch secures the normal outside air make-up to the CR. The hand switch, wiring, and affected components are safety grade, powered from safety grade electrical power supplies, and also provide a redundant capability to manually isolate CRACS should a failure occur in the automatic isolation logic. Therefore, Isolating the main CR can be accomplished by the normal compliment of MPS2's CR operating staff.

DNC's response to the NRC staff's request for additional information, dated March 6, 2007, described two types of MSLBs: (1) A large break, which results in a SIAS/CIAS/EBFAS from low RCS pressure, and (2) a smaller break which does not result in a SIAS/CIAS/EBFAS.

For a large MSLB, CRACS is designed to automatically isolate in response to the SIAS/CIAS/EBFAS. However, should a failure occur in the automatic isolation logic for CRACS, operators are directed by the emergency operating procedures to manually isolate CRACS.

For a smaller MSLB, where a SIAS/CIAS/EBFAS and automatic isolation of CRACS do not occur, EOP-2536, "Excess Steam Demand Event," will be entered, which directs the operators to manually isolate CRACS.

3.5.3 Verification of CR Isolation

Indications for the successful isolation of the CR include:

- "CRACS IN RECIRC" main control board annunciator
- Damper position indicating lights on the CRACS panels
- Fan running/not running indicator lights on the CRACS panels

All of the above indications are safety grade and powered from safety grade electrical power supplies. Therefore, the operators will be able to verify successful isolation of the CR.

3.5.3 ANSI/ANS 58.8-1994

In order to credit manual operator action(s), the time limit for taking the manual action(s) must allow for the operator to diagnose plant conditions and take the necessary action(s). For a steam line break and operating a single hand switch, ANSI/ANS 58.8-1994 specifies that the minimum time interval for diagnosis ($TI_{diagnosis}$) is 20 minutes, and the minimum time interval for subsequent operator action ($TI_{operator}$) is 6 minutes. This results in a total time interval from ANSI/ANS 58.8-1994 ($TI_{diagnosis} + TI_{operator}$) of 26 minutes, which is well within the 4 hour time limit proposed by DNC for completing this manual operator action. The 4 hour operator action time limit will therefore allow ample time for both operator diagnosis and operator action.

3.5.4 Human Performance Conclusion

Given that the actions to both diagnose the condition of a MSLB and take the appropriate actions are proceduralized and have been incorporated into operator training (including pass/fail criteria for this action during dynamic simulator exercises), the NRC staff finds that the assumed operator manual actions and completion times are acceptable.

3.6 Containment Sump pH

NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” notes that NUREG/CR-5732, "Iodine Chemical Forms in LWR Severe Accidents" indicates that the iodine entering the containment of a reactor coolant system during an accident would be composed of at least 95% cesium iodide (CsI). Upon deposition on interior surfaces and dissolution in the containment pool of a pressurized water reactor, the predominant form of the iodine would be the iodide ion (I^-). At pH less than 7, a large fraction of the iodide would be converted by irradiation into molecular iodine (I_2) and released into the containment atmosphere. If the pH were maintained above 7, however, the fraction of I^- converted to I_2 would be only about $3x10^{-4}$ (NUREG/CR-5732). During a LOCA, irradiation generates acids in the containment, principally hydrochloric acid (HCl) and nitric acid (HNO_3). In the absence of pH control, I_2 may be released during a LOCA as the acids lower the pH.

One way to minimize this release is to add an alkaline chemical capable of buffering the pH at a value above 7. At MPS2, trisodium phosphate (TSP) is stored as a solid in baskets in the containment sump. If a LOCA occurs, the rising borated water in the pool would dissolve the TSP, thus neutralizing acid in the pool water. The licensee provided the analysis used to determine the amount of TSP required to maintain a sump water pH of at least 7.1 for the 30-day period. This analysis included measurements of the pH in mixtures of borated water and small amounts of the TSP deployed in baskets in the MPS2 containment. After determining the relationship between boron concentration, TSP concentration, and pH, the licensee calculated the amount of TSP needed to raise the pH of the RCS water to 7.1 for bounding combinations of water volume and boron concentration. The amount of HNO_3 and HCl generated over the 30-day period was then estimated to determine the total amount of TSP required in the sump.

Hydroiodic acid (HI) is expected to be present in containments in small amounts (fewer than about 5 moles) and was not considered in the pH evaluations. Cesium hydroxide was also ignored in the pH evaluations, which adds conservatism since it is an alkaline species and therefore raises the pH. The NRC staff reviewed these assumptions and considers them to be appropriate.

3.6.1 Boric Acid from the RCS

Based on pH measurements for boric acid solutions, the licensee determined that the solution of borated water (from the reactor, pressurizer, RWST, safety injection tanks, and boric acid storage tanks) has a nominal room-temperature (77 °F) pH of about 4.6 for the maximum boron concentration (i.e., minimum pH) of about 2700 parts per million (ppm). Hence, this is the lowest pH expected for the pool when the reactor coolant is released at the onset of a LOCA. This is comparable to the approximate value of 4.9 calculated by the NRC staff using standard acid-base equations.

3.6.2 Nitric Acid

HNO_3 would be produced by the irradiation of water and air following a LOCA. The amount of acid produced is proportional to the time-integrated dose rate for gamma and beta radiation. The licensee calculated a total of 598 moles generated, based on a production rate of $7.3x10^{-6}$ moles HNO_3 per liter per megarad (Mrad), which is consistent with NUREG/CR-5950.

3.6.3 Hydrochloric Acid

HCl would be produced from the irradiation of chloride-containing electrical insulation following a LOCA. The amount of acid generated is proportional to the amount of beta and gamma radiation absorbed by the insulation. The licensee calculated a total of 3107 moles generated. This was based on a production rate of 4.6×10^{-4} moles of HCl per pound of insulation per Mrad, consistent with NUREG/CR-5950.

3.6.4 TSP pH Buffer

The containment sump at MPS2 contains five baskets of solid TSP in the hydrated form ($Na_3PO_4 \bullet 12H_2O \bullet \frac{1}{4}NaOH$), which would dissolve as the water level in the sump rises during a postulated LOCA. Using the amount of boric acid, HNO_3, and HCl introduced into the sump, the licensee calculated the amount of TSP required to maintain the sump pH at or above 7.1 for 30 days following the postulated LOCA. The MPS2 technical specifications specify a minimum of 282 cubic feet of TSP (in the hydrated form described by the formula above).

3.6.5 Containment Pool pH Calculation

The licensee's calculations varied the coolant volume and boron concentration in the RCS components according to the range of allowable values in order to find the minimum sump pH (boric acid alone). The licensee also calculated the maximum sump pH based on the initial dissolution of the TSP. The NRC staff reviewed the licensee's assumptions and methodology, performed confirmatory calculations, and used an alternative source of pH values for boric acid/TSP mixtures to evaluate the licensee's conclusions. The result of the NRC staff's evaluation was consistent with the licensee's conclusion that the sump pH would remain at or above 7 for 30 days following a postulated LOCA. By maintaining the pH above 7, the fraction of radioactive iodine released into the containment atmosphere following a LOCA will be minimized.

3.7 <u>Summary</u>

As described above, the staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of DBAs with full implementation of an AST at MPS2. The staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 above. The staff compared the doses estimated by the licensee to the applicable criteria identified in Section 2.0. The staff also finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and CR doses will comply with these criteria. The staff further finds reasonable assurance that the DBA radiological consequences analysis for MPS2, as modified by this license amendment, will continue to provide sufficient safety margins with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameters. Therefore, the proposed license amendment is acceptable with respect to the radiological consequences of DBAs.

This licensing action is considered a full implementation of the AST. With this approval, the previous AST in the MPS2 design-basis is superseded by the AST proposed by the licensee. The previous offsite and CR accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR 50.67, or fractions thereof, as defined in Regulatory Guide 1.183. All future radiological accident analyses performed to show compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as defined in the MPS2 design-basis, and modified by the present amendment.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Connecticut State official was notified of the proposed issuance of the amendment. The Connecticut State official provided no comments.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (71 FR 51226). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Attachment: Tables 1 - 9

Table 1
MPS2 Radiological Consequences Expressed as TEDE [1]
(rem)

Design-Basis Accidents	EAB [2]	LPZ [3]	Control Room
Loss of Coolant Accident	2.9	1.7	3.0
Dose Criteria	25	25	5.0
Fuel Handling Accident	1.5	0.2	3.1
Dose Criteria	6.3	6.3	5.0
Cask Tip Accident	0.5	0.05	0.25
Dose Criteria	6.3	6.3	5.0
Steam generator tube rupture [4]	1.4	0.18	4.5
Dose criteria	25	25	5.0
Steam generator tube rupture [5]	1.2	0.16	4.5
Dose criteria	2.5	2.5	5.0
Main steamline break accident [4][6]	0.091	0.028	2.6
Dose criteria	25	25	5.0
Main steamline break accident [5][6]	0.16	0.054	3.8
Dose criteria	2.5	2.5	5.0
Main steamline break accident [4][7]	0.091	0.029	4.0
Dose criteria	25	25	5.0
Main steamline break accident [5][7]	0.16	0.054	4.7
Dose criteria	2.5	2.5	5.0
CREA[8] Accident			
Containment	0.54	0.47	1.6
Secondary side	0.79	0.18	3.9
Dose criteria	6.3	6.3	5.0

[1] Total effective dose equivalent
[2] Exclusion area boundary
[3] Low population zone
[4] Pre-accident iodine spike
[5] Concurrent iodine spike
[6] Enclosure Building
[7] Turbine Building
[8] Control Rod Ejection Accident

Table 2
Control Room Atmospheric Dispersion Factors

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

Table 3
Offsite Atmospheric Dispersion Factors (sec/m^3)

Receptor/ Source Location / Duration		χ/Q (sec/m^3)
EAB (0 - 720 hours)		
Millstone Stack release (includes fumigation)		1.00E-04
Other Release Points		3.66E-04
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		
	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

Table 4
MPS2 Control Room Data and Assumptions

CR effective volume	35,656 ft^3
Normal CR intake flow rate prior to isolation	800 cfm
Unfiltered inleakage	200 cfm
Emergency Ventilation System Recirculation flow rate	2250 cfm
Response time for CR to Isolate upon receipt of Control Room Ventilation Radiation Monitor Alarm Signal or Enclosure Building Filtration Actuation/SI signal	20 seconds
Manual CR isolation time for MSLB	4 hours
Time allotted for operator action to align CREV after isolation to filtered, recirculation mode	1 hour
Total time allotted to place CREV in service (summation of the 2 preceding time intervals)	1 hr and 20 seconds
Filter Efficiencies for CREV	90 percent elemental 90 percent aerosol 70 percent organic
Containment wall thickness	3 ft 9 in concrete
Containment dome thickness	3 ft 3 in concrete
Control building wall thickness	2 ft concrete
Control room ceiling/roof thickness	2 ft concrete
CR occupancy factors	
0 - 24 hours	1.0
24 - 96 hours	0.6
96 - 720 hours	0.4
Breathing rate for CR dose analyses	3.5E-04 m^3/sec

Table 5
MPS2 Data and Assumptions for the LOCA

Containment free air volume	1.899E+06 ft^3
Containment leak rate	0.5 percent weight percent per day (L_a)
Containment bypass leak rate	$0.014L_a$
Containment leak rate reduction	50 percent after 24 hours
Secondary containment drawdown time	110 seconds
Iodine chemical form in containment atmosphere	95 percent cesium iodide 4.85 percent elemental iodine 0.15 percent organic iodine
Iodine chemical form in the sump and RWST	97 percent elemental 3 percent organic
Containment sump pH	≥ 7
EBFS filter efficiency	70 percent all iodines and particulates
Containment Spray effective operation period	
elemental	75 seconds - 3.03 hours
particulate	75 seconds - 3.23 hours
Elemental iodine removal coefficient	20 per hour
Particulate iodine removal coefficient	
DF < 50:	6.42 per hour
DF > 50:	not credited
Containment Sprayed Volume Percentage	35.4 percent
Start time for ECCS leakage	27.5 minutes
ECCS leakage outside containment	24 gph
RWST minimum volume	282,200 gallons
RWST maximum volume	475,000 gallons

Table 6
MPS2 Data and Assumptions for the FHA and Cask Tip Accident

Fuel clad damage	
FHA	All rods in 1 assembly
Cask Tip	All rods in 184 assemblies decayed 1 year and 1376 assemblies decayed 5 years
Gap fractions	
I-131	12 percent
Kr-85	30 percent
Other noble gases	10 percent
Other halogens	10 percent
Pool DF	
Noble gases	1
Iodines	200 (effective DF)
Release point	Enclosure Building / Containment Ground
Decay time	
FHA	100 hours
Cask Tip	1 year or 5 years
Radial peaking factor	1.83
Duration of release	2 hours
Control room ventilation timing:	
T= 0 seconds	Normal CR unfiltered intake flow - 800 cfm
T= 20 seconds	Control room isolates Intake flow - 0 cfm - neutral condition Assumed unfiltered inleakage - 200 cfm
T = 1 hour 20 seconds	CREV filtered intake flow - 0 cfm Unfiltered inleakage - 200 cfm CREV filtered recirculation flow - 2250 cfm

Table 7
MPS2 Data and Assumptions for the SGTR Accident

Primary-to-secondary leak rate TS limit	150 gpd (to intact SGs)
Secondary iodine TS limit	0.1 µCi/gm DE I-131
RCS TS limit for normal operation	
Gross gamma	100/ E_{bar}
Iodine	1.0 µCi/gm DE I-131
RCS TS limit for pre-accident iodine spike	60 µCi/gm DE I-131
Coincident spike appearance rate multiplier	335
Iodine spike duration	8 hours
LOOP	Coincident with release
Release points	SG ADVs/MSSVs
Secure release from affected SG	1 hour
Secure release from intact SGs	17 hours
Chemical form of iodine released from SGs	Elemental 97 percent Organic 3 percent
Iodine PC	100
Moisture carryover in intact SGs	0.4 percent
Duration of release to environment	
Intact SGs	0 - 17 hours
Affected SG	0 - 2 hours
Initial SG mass	280,000 lbm / SG
CR ventilation timing	
Pre-accident Spike	20 seconds CR isolation 1 hour, 20 seconds CREV initiated
Concurrent Spike	10 minutes, 20 seconds CR isolation 1 hour, 10 minutes, 20 seconds CREV initiated

Table 8
MPS2 Data and Assumptions for the MSLB Accident

Primary-to-secondary leak rate TS limit	150 gpd (to intact SGs)
Secondary iodine TS limit	0.1 µCi/gm DE I-131
RCS TS limit for normal operation	
Gross gamma	100/ E_{bar}
Iodine	1.0 µCi/gm DE I-131
RCS TS limit for pre-accident iodine spike	60 µCi/gm DE I-131
Coincident spike appearance rate multiplier	500
Iodine spike duration	8 hours
LOOP	Concurrent
Release points:	
Affected SG	Turbine building
Intact SG	ADVs
Iodine PC for intact SGs	100
Moisture carryover in intact SGs	0.4 percent
Primary-to-secondary leakage	
Affected SG	150 gpd
Total	1 gpm
SG liquid mass	91,092 lbm min. 248,981 lbm max.
CR ventilation timing	
Enclosure & Turbine Bldg Releases	4 hours: CR isolation on operator action 5 hours: CREV initiation
Containment Release	140 seconds: CR isolation on SI signal 1 hour, 140 seconds: CREV initiation
Duration of SG release:	
Affected SG	36 hours
Intact SG	16 hours
Steam release from affected SG	
Initial inventory	3.08E+05 lbm/min (0 - 750 sec)
Primary-to-secondary leak	0.868 lbm/min (0 - 36 hrs) 0.868 lbm/min (0 - 720 hrs) Enclosure Bldg
Integrated Steam Release from intact SG	
0 - 20 seconds	54,334 lbm
20 - 800 seconds	54,334 lbm
800 - 10,000 seconds	534,459.42 lbm
10,000 - 60,000 seconds	1,979,171.92 lbm

Table 9
MPS2 Data and Assumptions for the CREA

Containment free air volume	1,899,000 ft^3
Fraction of fuel clad failure	0.115
Fraction of core inventory in gap	
Noble gasses	0.1
Iodine	0.1
SI signal initiated after CREA	1 minute
Chemical form of iodine released from the SG to the environment	3 percent organic iodide 97 percent elemental iodine
Total primary-to-secondary leakage through all SGs	300 gpd
Duration of steam releases	60,000 seconds
CR ventilation timing	
Time for CR isolation	105 seconds

Millstone Power Station, Unit No. 2

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