September 15,2006

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. 3 - ISSUANCE OF AMENDMENT RE: ALTERNATE SOURCE TERM (TAC NO. MC3333)

Dear Mr. Christian:

The Commission has issued the enclosed Amendment No. 232 to Facility Operating License No. NPF-49 for the Millstone Power Station, Unit No. 3. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated May 27, 2004, as supplemented by letters dated September 27 and October 20, 2004; March 23, 2005; and January 30 and May 25, 2006.

This amendment revises the TSs to incorporate a full-scope application of an alternate source term methodology in accordance with Title 10 of the *Code of Federal Regulations*, Section 50.67.

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/

Victor Nerses, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-423

Enclosures:
1. Amendment No. 232 to License No. NPF-49
2. Safety Evaluation

cc w/encls: See next page

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DOMINION NUCLEAR CONNECTICUT, INC.

DOCKET NO. 50-423

MILLSTONE POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 232
License No. NPF-49

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

   A. The application for amendment filed by Dominion Nuclear Connecticut, Inc. (the licensee) dated May 27, 2004, as supplemented by letters dated September 27 and October 20, 2004; March 23, 2005; and January 30 and May 25, 2006, 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. NPF-49 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. 232, 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.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Brooke D. Poole, Acting Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 15, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 232

FACILITY OPERATING LICENSE NO. NPF-49

DOCKET NO. 50-423

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
x	x
xiv	xiv
1-2	1-2
3/4 7-18	3/4 7-18
3/4 7-19	3/4 7-19
6-17	6-17

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 232

TO FACILITY OPERATING LICENSE NO. NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.

MILLSTONE POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

By letter dated May 27, 2004, as supplemented by letters dated September 27 and October 20, 2004; March 23, 2005; and January 30 and May 25, 2006, Dominion Nuclear Connecticut, Inc. (DNC or the licensee) submitted an application requesting a license amendment for Millstone Power Station, Unit No. 3 (MPS3). The application provides the Technical Specification (TS) changes and evaluations of the radiological consequences of design-basis accidents (DBAs) for implementation of a full-scope alternate source term (AST), pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67, "Accident Source Term," and using the methodology described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 1).

The supplements dated September 27 and October 20, 2004; March 23, 2005; and January 30 and May 25, 2006, 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 consideration determination as published in the *Federal Register* on September 14, 2004 (69 FR 55468).

2.0 REGULATORY EVALUATION

In the past, power reactor licensees have typically used U.S. Atomic Energy Commission Technical Information Document TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962, as the basis for DBA analysis source terms. The power reactor siting regulation, which contains offsite dose limits in terms of whole-body and thyroid dose, 10 CFR Part 100, Section 11 (10 CFR 100.11), "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," makes reference to TID-14844.

Since the publication of TID-14844, significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. Many of these insights developed out of the major research efforts started by the Nuclear Regulatory Commission (NRC or the Commission) and the nuclear industry after the accident at Three Mile Island. In 1995, the NRC published NUREG-1465, "Accident Source

Terms for Light-Water Nuclear Power Plants," (Reference 2) that utilized this research to provide more physically-based estimates of the accident source term that could be applied to the design of future light-water power reactors. These revised source terms are described in terms of radionuclide composition and magnitude, physical and chemical form, and timing of release.

In December 1999, the NRC promulgated 10 CFR 50.67, which provides a mechanism for licensed power reactors to replace the traditional accident source term used in their DBA analyses with an AST. Regulatory guidance for the implementation of these ASTs is provided in RG 1.183. Section 50.67 requires a licensee seeking to use an AST to apply for a license amendment, and requires that the application contain an evaluation of the consequences of applicable DBAs previously analyzed in the safety analysis report.

DNC's application of May 27, 2004, as supplemented by letters dated September 27 and October 20, 2004; March 23, 2005; and January 30 and May 25, 2006, addresses the requirements in proposing to use the AST described in RG 1.183 as the source term used in the evaluation of the radiological consequences of selected DBAs at MPS3.

The NRC staff evaluated the radiological consequences of applicable DBAs for implementation of the AST methodology at MPS3, 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. Except where the licensee has proposed a suitable alternative, the NRC staff used the following regulations, regulatory guidance, and standards in its review:

- 10 CFR 50.36,"Technical specifications"
- 10 CFR 50.67
- 10 CFR Part 50, Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," GDC 19, "Control Room"; GDC 60,"Control of Releases of Radioactive Materials to the Environment"
- RG 1.145,"Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants", Rev. 1
- RG 1.183 (Reference 1)
- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants", Rev. 0
- RG 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors", Rev. 0 (May 2003)
- RG 1.23, "Onsite Meteorological Programs", Rev. 0

- NUREG-0800, Rev. 0 (July 2000) "Standard Review Plan:" Section 2.3.4, "Short-Term Diffusion Estimates for Accidental Atmospheric Releases (1996 draft version);" Section 6.4, "Control Room Habitability Systems (1996 draft version);" Section 15.0-1, "Radiological Consequence Analyses Using Alternative Source Term (Rev. 0, July 2000);" and Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment (1996 draft version)."

The NRC staff considered the impact of the proposed changes on the previously-analyzed DBA radiological consequence analysis, and the acceptability of the revised analysis results.

3.0 TECHNICAL EVALUATION

3.1 Radiological Consequences of DBAs

The current DNC DBAs analyzed for the radiological consequences at the EAB, LPZ, and in the CR in Section 15, "Accident Analyses" of the DNC, MPS3 Updated Final Safety Analysis Report (UFSAR) include the following nine events:

10. Main Steamline Break (MSLB)
11. Locked Rotor Accident (LRA)
12. Rod Control Cluster Assembly (RCCA) Rod Ejection Accident (REA)
13. Small-Line Loss-of-Coolant Accident (LOCA) Outside Containment
14. Steam Generator Tube Rupture (SGTR)
15. LOCA
16. Fuel Handling Accident (FHA)
17. Waste Gas System Failure
18. Radioactive Liquid Waste System Leak or Failure (Atmospheric Release)

The DBAs addressed in RG 1.183 were selected from accidents that may involve damage to irradiated fuel. RG 1.183 does not address DBAs with radiological consequences based on TS reactor or secondary coolant-specific activities only. The inclusion, or exclusion, of a particular DBA in RG 1.183 should not be interpreted as indicating that an analysis of that DBA is required or not required. Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST. The proposed change, for the full implementation of the AST methodology for MPS3, does not affect the evaluation of the radiological consequences for "Failure of Small Lines Carrying Primary Coolant Outside Containment," since there is no fuel damage postulated for this event. In a letter dated May 25, 2006, the licensee stated that it will retain the existing discussion of this accident in the UFSAR.

The licensee will retain the radiological consequences of the waste gas system failure and the radioactive liquid waste system leak or failure (atmospheric release) in Chapter 11 of the MPS3 UFSAR and will continue to express the doses as whole-body and thyroid with the acceptance criteria of 500 milirem whole-body. The licensee will not convert the doses for these system failure events to TEDE. The evaluation of these system failures is generally based on the release of the maximum quantity of radioactive material allowed to be stored in the respective systems as governed by TSs. The proposed change to an AST does not affect the TS pertinent to these systems, therefore no re-analysis is necessary. The NRC staff agrees that these evaluations are appropriately retained in Chapter 11 of the MPS3 UFSAR.

A full implementation of the AST (as defined in Section 1.2.1 of RG 1.183) is proposed for MPS3. Therefore, to support the licensing and plant operation changes discussed in Section 2.0 of the license amendment request, the licensee analyzed the following accidents employing the AST as described in RG 1.183.

19. LOCA
20. FHA
21. SGTR Accident
22. MSLB Accident
23. LRA
24. REA

The licensee performed dose calculations 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 MPS3 CR operator were evaluated for the duration of the accident. DNC performed all the radiological consequence calculations for the AST with the RADTRAD-NAI computer code. 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 six events 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.

3.1.1 LOCA

The radiological consequence design-basis LOCA 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 deterministic 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 analysis, the licensee included dose contributions from the following sources:

- Containment leakage plume
- ECCS component leakage
- reactor water storage tank (RWST) vent releases
- Shine from containment and the plume
- Shine from the CR filter loading

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 code. The inventory, consisting of 66 isotopes at the end of fuel cycle curie levels, 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, as is appropriate for the AST evaluation. The use of ORIGEN and DCFs from FGR 11 and FGR 12 is in accordance with RG 1.183 guidance and 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. Credit for the use of containment sprays for elemental and particulate iodine removal by the quench spray system (QSS) was approved in Amendment No. 211, dated September 16, 2002 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML022470399), and November 25, 2002 (ADAMS Accession No. ML023290568). In Amendment No. 211, the staff approved a QSS coverage value of 50.27 percent and a QSS effective initiation time of 70.2 seconds. The proposed MPS3 AST analysis assumes that the percentage of containment that is covered by quench spray is 49.63 percent and that the QSS becomes effective at 72.5 seconds (0.02014 hours) post-LOCA. In addition, the AST analysis proposes to credit the recirculation spray system (RSS) for containment iodine removal at 14 minutes (0.2333 hours) post-LOCA, thereby increasing the sprayed coverage to 64.5 percent during the time when both spray systems are operating. The mixing rate during spray operation is assumed to be two turnovers of the unsprayed volume per hour, which is consistent with the value accepted by the staff in Amendment No. 211.

In a facsimile dated September 22, 2004, the NRC forwarded a request for additional information (RAI) concerning the analyses performed to determine the containment spray volume coverage fractions used in the AST analyses. The licensee responded to this request in a letter dated October 20, 2004.

The containment free volume of 2.32E+06 cubic feet (ft^3), used in the current licensing analysis of record (AOR) to determine the spray coverage fraction, is within the range of the containment free volume, as calculated by the licensee in Calculation 12179-ES-227, Revision 0, 1980. The value used in the AST analysis is based on the maximum containment free volume of 2.35E+06 ft^3, with the additional volume conservatively considered to be part of the unsprayed volume. The containment spray volume coverage percentage was conservatively reduced from 50.27 percent to 49.63 percent during the period of time when only the QSS is active. The methodology used to obtain the QSS volume coverage percentage was not changed from the previous AOR, based on Licensee Calculation US(B)-341. The change in the QSS coverage percentage is only attributable to the increase in the assumed total containment free volume.

The AST analysis credits the increase in the containment spray volume coverage fraction during the period of time when both the QSS and the RSS are functioning. After the QSS is secured, even though the RSS is still working, no credit for the containment sprays is taken. The methodology used to determine the containment spray volume coverage fraction from the RSS is similar to the method described in US(B)-341. In the determination of the RSS spray coverage, a conservative evaluation was performed by the licensee. A 50-percent spray coverage factor was used for the containment annulus region, which included reduced volume because of equipment located in this region. The combined (QSS plus RSS) containment spray volume used in the AST analysis was about 7 percent lower than the calculated volume. This added additional conservatism and reduced the containment spray volume coverage percentage from 65 percent to 64.5 percent.

The containment spray volume coverage fractions for the QSS and the combined QSS plus RSS periods of operations are based on conservative evaluations of the total containment spray volume and the coverage fractions are, therefore, acceptable to the NRC staff for AST analyses.

The licensee applied calculated spray removal rates until the QSS is secured at 7480 seconds (2.078 hours). At that time, further iodine removal due to sprays is not credited by the licensee even though the RSS remains operating. For the time period during which sprays are assumed operating, the licensee calculated an elemental iodine decontamination factor (DF) of 79. The licensee calculated that a particulate iodine DF of 49.5 would be attained at 6840 seconds (1.9 hours), at which time the calculated particulate removal rate was reduced by a factor of 10, in accordance with Standard Review Plan (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 radioactivity in the unsprayed region of the containment by natural deposition. The licensee used the Powers model as incorporated into the RADTRAD computer code. The licensee conservatively credited 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 does not double-count iodine removal in the containment by limiting credit for natural removal to the unsprayed regions only.

3.1.1.3 Secondary Containment Bypass Release Pathways

The licensee's analysis supporting this amendment request addresses a plant-specific issue of unfiltered post-LOCA releases due to damper bypass and duct leakage from the plant ventilation system that was described and approved in Amendment No. 211. Amendment No. 211 identified potential release pathways from the secondary containment to the environment that could bypass the supplementary leakage collection and release system (SLCRS) filter following a DBA due to non-nuclear safety (NNS) grade exhaust fan operation after the accident. Amendment No. 211 also approved an operator action that would manually trip the breakers on selected NNS exhaust fans at 1 hour and 20 minutes post-LOCA. This operator action is only credited in the CR habitability analysis. In the AST analysis, the licensee did not change the licensing basis for the post-accident operation of the SLCRS as described and approved in Amendment No. 211 and, therefore, the assumptions related to its operation remain acceptable to the NRC staff.

3.1.1.4 Containment Leakage

The total containment leakage (L_a) for MPS3 consists of filtered and bypass leakage and, as governed by the TSs, is 0.3 weight percent per day. The entire containment leak rate bypasses the secondary containment until the SLCRS drawdown is effective at 2 minutes post-LOCA. After SLCRS drawdown, the bypass leak rate is assumed to be reduced by a factor of 0.06 to 0.018 weight percent per day. The licensee assumes this bypass leak rate is released unfiltered at ground level directly from containment. The containment leak rate L_a, and the bypass leak rate are reduced by one-half at 24 hours for offsite calculations, and at 1 hour for CR calculations. This assumption, of a reduction in the containment leak rate by 50 percent after 1 hour for the CR habitability analysis, was used in calculations supporting Amendment No. 59, issued January 25, 1991 (Reference 4) which eliminated the post-LOCA negative containment pressure requirement. This assumption was also referenced in the description of calculations provided as supplemental information (Reference 5) supporting Amendment No. 211 (Reference 6) which changed the licensing basis for the post-accident operation of the SLCRS. The assumption of a 50-percent reduction in containment leakage after 1 hour is based on the fact that the MPS3 post-LOCA containment pressure is rapidly reduced compared to typical pressurized water reactors (PWRs) because of its original design as a negative pressure containment.

3.1.1.5 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 (SI) 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. 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 suggested 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 TS 6.8.4a, "Primary Coolant Sources Outside Containment Program Manual," which calculates the maximum allowable leakage as 4780 cubic centimeters per hour (cc/hr). 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 10,000 cc/hr for the evaluation of the ECCS leakage contribution to the LOCA dose.

The licensee assumed that the leakage of recirculating sump fluids commences at 640 seconds, which is the earliest time that the recirculation of contaminated fluids would begin. The licensee calculated a flashing fraction of 0.03, which corresponds to an assumed maximum containment sump temperature of 240 EF. However, in following the guidance of RG 1.183, the licensee conservatively used a flashing fraction of 0.1 for the ECCS leakage calculation for the duration of the event. As a result, 10 percent of the entrained iodine activity in the ECCS leakage effluent is assumed to be released to the atmosphere of the surrounding auxiliary building. 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.

3.1.1.6 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 valving 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 used RADTRAD-NAI to model 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 releases from 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 Amendment 176, issued December 1, 1999 (Reference 7), to calculate times, flow rates, and volumes for each identified pathway.

Using the methodology approved in Amendment 176, the licensee based the time for contaminated sump water to reach the RWST 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 licensee calculated an RWST vent airflow rate using the ideal gas law. The expected RWST volumetric changes were determined based on conservative estimates of the rise in air temperature within the RWST due to solar heating. The licensee performed a detailed analysis to determine the partition coefficient (PC) applicable to the iodines in the RWST water. The licensee calculated a DF of 450 for the release of iodines from the RWST as a result of back leakage. However, for conservatism, the licensee used a DF of 100 for the evaluation of the dose contribution from RWST back leakage.

3.1.1.7 LOCA Conclusion

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 provided in 10 CFR 50.67 and guidelines provided in the accident dose criteria specified in SRP 15.0.1. The NRC staff's review found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this safety evaluation (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 staff performed independent confirmatory dose evaluations to ensure a thorough understanding of the licensee's methods. The 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

This accident analysis postulates that a spent fuel assembly is dropped during fuel handling and strikes an adjacent assembly during the fall. All of the fuel rods in the dropped assembly and 50 fuel rods in the struck 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 affected assemblies are assumed to be those with the highest inventory of fission products of the 193 assemblies in the core. Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod clad during normal power operations. 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, a DF of 200 for radioiodines, and retention of all aerosol and particulate fission products. As suggested 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 219, issued March 17, 2004 (Reference 8), for selective implementation of the AST.

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

- CR atmospheric dispersion factors (χ/Qs)
- CR inleakage assumptions
- CR filtration efficiency
- CR Envelope Pressurization System (CREPS) modeling

The evaluation of the DNC atmospheric dispersion factors is discussed in SE Section 3.1.8.

The licensee made several changes to the assumptions governing CR habitability that prompted requests for additional information from the NRC staff. The changes to the CR habitability assumptions are discussed further in Section 3.1.7, "Control Room Habitability," of this SE. For the FHA, the licensee assumes that it will take 10 seconds for the CR to isolate following detection of released activity by the MPS3 CR air inlet detectors. In an isolated or neutral pressure condition, the licensee assumes an unfiltered inleakage of 350 cubic feet per minute (cfm). This assumption provides a considerable margin above measured test results, which indicate unfiltered inleakage for the neutral condition of less than 100 cfm. The licensee does not credit the operation of the CREPS in any of the AST DBA dose analyses and by this amendment is removing the associated TS for this system. However, the licensee is not physically removing the CREPS and, therefore, in the time sequence of the FHA analysis, the licensee has allotted time for the operation of the CREPS without crediting pressurization or its inleakage reducing benefit.

For the FHA, the licensee assumes that the CR will experience 10 seconds of normal ventilation intake flow at 1595 cfm prior to CR isolation. After CR isolation, the licensee assumes a period of 101 minutes of unfiltered inleakage at 350 cfm. The 101-minute period assumes a neutral pressure condition that accounts for the delay to the activation of the safety-related control room emergency ventilation system (CREVS), including 1 minute for CREPS actuation, 60 minutes for CREPS operation and 40 minutes for the manual alignment of the CREVS. This assumption is very conservative in that it assumes an unfiltered inleakage of 350 cfm persists for a 61 minute time period allotted for CREPS operation, in addition to the 40 minutes allotted for the manual alignment of the CREVS. During the 61 minute time period allotted for CREPS operation, there is no credit taken for its inleakage reduction benefit. At T= 1.685 hours (101 minutes, 10 seconds), the licensee assumes that the CREVS is actuated providing a filtered intake flow of 230 cfm. The unfiltered inleakage is assumed to be reduced to 100 cfm due to the operation of the CREVS.

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 guidelines provided in 10 CFR 50.67 and accident dose criteria specified in SRP 15.0.1. The NRC staff's review 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 staff performed independent confirmatory dose evaluations to ensure a thorough understanding of the licensee's methods. The 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 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). For this accident scenario, a loss of off-site power (LOOP) is assumed to occur concurrently with the tube rupture. Within the first few minutes of the accident, the CR isolates, the reactor trips, and SI is actuated. 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 on the affected SG is assumed to open to control SG pressure at the beginning of the event. After operator action is credited to close the affected SG ADV, the same ADV is assumed to fail fully open. The affected SG discharges steam to the environment for 2946 seconds (0.8183 hours) until the generator is manually isolated a second time by closure of the SG atmospheric dump block valve. Break flow into the affected SG continues until 5596 seconds (1.554 hours), at which time the RCS is at a lower pressure than the secondary system. Additional releases from the affected SG are modeled from 2-8 hours to complete depressurization of the SG early in the event. Depressurization of the SG is necessary to allow residual heat removal system (RHRS) cooling.

The licensee modeled the three unaffected SGs as one volume, referred to in the analysis as the intact SG. The licensee evaluated the dose consequences from discharges of steam from the intact SG for a period of 18 hours, until the primary system has cooled sufficiently to allow an alignment to the RHRS. As a result of the assumption of a concurrent LOOP, the 18-hour period includes an additional 7 hours of steaming required to reduce the system heat load to the point where the RHRS can remove all the decay heat using only safety grade equipment. After a period of 18 hours, the RHRS 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 releases from the intact SGs are 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 the TSs. 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

TSs for a spiking condition. For MPS3, the maximum iodine concentration allowed by the TSs as a result of an iodine spike is 60 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 MPS3, 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 analysis basis 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's analyses are discussed below.

3.1.3.1 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. In the DNC analysis for MPS3, break flow is terminated after 1554 hours. 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. 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 be released to the environment in direct proportion to the steaming rate and in inverse proportion to the applicable PC. 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 all non-noble gas isotopes. 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.

3.1.3.2 Releases from the Intact SGs

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 SGs by the leak rate limiting condition for operation (LCO) (1 gpm) specified in the TS. All radionuclides in the primary coolant leaking into the intact SGs 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 all non-noble gas isotopes. Therefore, 1 percent of the iodines and 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 SGs are released to the environment without reduction or mitigation. Releases were assumed to continue from the intact SGs for a period of 18 hours until the primary system cools to below 350 EF and the RHRS is able to remove 100 percent of decay heat with no requirement for steaming to augment cooldown. The 18-hour steaming period is based on the time necessary to cool down crediting safety grade equipment only.

3.1.3.3 SGTR Accident Conclusion

The licensee evaluated the radiological consequences resulting from the postulated SGTR accident and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident dose criteria specified in SRP 15.0.1. The NRC staff's review 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 7 and the licensee's calculated dose results are given in Table 1. The staff performed independent confirmatory dose evaluations to ensure a thorough understanding of the licensee's methods. The 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.

3.1.4 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 the largest main steamline outside containment. The radiological consequences of an MSLB outside containment will bound the radiological consequences of a break inside containment. Therefore, only the MSLB outside of containment is considered with regard to the radiological consequences.

The licensee's evaluation indicates that no fuel damage is predicted as a result of an MSLB accident. Therefore, consistent with the current licensing analysis basis and RG 1.183, the licensee performed the MSLB accident 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 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 described in SE Section 3.1.3 for the SGTR with the following exception. For the MSLB accident, 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 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 MSLB accident begins with a break in one of the main steamlines 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. In order to maximize the CR dose, the licensee assumed that the steamline break occurs in the turbine building. The affected SG is assumed to release steam for 55.2 hours, which is the time required for the RCS to be cooled down to 212 EF. The 55.2-hour steaming period is based on the time necessary to cool down to 212 EF, crediting safety grade equipment only. The licensee has conservatively determined that dry out of the affected SG will occur 56.3 seconds after the MSLB.

The licensee evaluated the 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. The release from the intact SGs continues for 18 hours through the ADVs until the RHRS can fully remove decay heat.

The licensee assumed that the source term resulting from the radionuclides in the primary system coolant and from the iodine spiking in the primary system is transported to the SGs by the leak-rate LCO (1 gpm) specified in the TS. The maximum amount of primary-to-secondary leakage allowed by the TS to any one SG is 500 gallons per day (gpd). The licensee conservatively assigned this leakage (500 gpd or 0.35 gallons per minute (gpm)) to the affected SG to maximize the calculated dose consequence.

For the affected SG, DNC assumed the release passes directly into the turbine 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. The DNC analysis assumes the release into the turbine building is exhausted to the environment and subsequently transported from the environment into the CR assuming conservative atmospheric dispersion factors.

The licensee assumed that during the first 56.3 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 near instantaneous release of the secondary coolant from the affected SG represents a significant contribution to the total dose from an MSLB, since the inventory is evaluated at the secondary side TS limit of 0.1 µCi/gm DE I-131. The licensee conservatively assumed that during the first 55.2 hours, primary coolant leaks into the affected SG at the rate of 500 gpd (0.35 gpm) directly releasing all of the coolant activity to the environment. This release is assumed to continue for 55.2 hours, until the RCS has cooled to below 212 EF, at which time the release from this pathway terminates. 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.

The licensee assigned the remainder of the 1 gpm primary-to-secondary side leakage (0.65 gpm) to two of the three intact SGs. The licensee assumed that this leakage continues for 18 hours, until shutdown cooling is initiated and credited for decay heat removal. The licensee's analysis assumes that the third intact SG has a failed closed atmospheric dump valve. This assumption reduces the holdup volume to that of two SGs instead of three. However, in order to maximize the release rate from this pathway, the licensee did not reduce the assumed steaming rate.

In order to ensure proper accounting of gross gamma, iodine and noble gas releases from the intact SGs, the licensee evaluated all the significant nuclide transport models for the MSLB accident. The licensee evaluated the release of the gross gamma activity from the primary coolant, at the TS limit of $100/E_{bar}$, leaking into the intact SG volume at a primary-to-secondary leak rate of 0.65 gpm. 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 PC of 100 results in 1 percent of the particulates and iodines in the SG bulk liquid being released to the environment at the steaming rate. 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.

The licensee evaluated the radiological consequences resulting from the postulated MSLB accident and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident dose criteria specified in SRP 15.0.1. The NRC 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 8 and the licensee's calculated dose results are given in Table 1. The staff performed independent confirmatory dose evaluations to ensure a thorough understanding of the licensee's methods. The 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.

3.1.5 Primary Coolant Pump Locked Rotor Accident

The accident considered begins with the instantaneous seizure of a reactor coolant pump rotor, which causes a rapid reduction in the flow through the affected RCS loop. The sudden decrease in core coolant flow while the reactor is at power causes a degradation of core heat transfer, resulting in localized temperature and pressure changes in the core. As a result, the licensee assumes that fuel damage occurs due to a departure from nucleate boiling. Activity from the fuel damage is transported to the secondary side due to primary-to-secondary side leakage evaluated at the TS limit. It is assumed that the LRA does not cause an increase in the magnitude of the pre-existing primary-to-secondary leakage.

The licensee incorporated the assumption of a coincident LOOP into the analysis. This results in a release through an assumed stuck open SG ADV and additional releases from the intact SGs. The stuck open ADV represents the assumed single active failure. The licensee stated that, consistent with the current licensing basis, operator action to close the stuck open ADV is credited after 20 minutes.

As a result of the LRA, the licensee has determined that 7 percent of the fuel inventory gap activity would be released to the RCS. The RCS source term is assumed to be transported to the secondary side at the TS primary-to-secondary leakage limits of 1 gpm total with 500 gpd (0.35 gpm) assigned to the affected SG. The licensee assumed that the release from the affected SG continues for 20 minutes, at which point operator action is credited for the isolation of that release pathway.

The licensee assumed that the balance of the total 1 gpm TS limit (0.65 gpm) is released from the intact SGs over the course of 18 hours until shutdown cooling can be implemented to fully remove decay heat crediting only safety grade equipment. At this point, the release from the intact SGs is terminated when the operator closes the ADVs.

The licensee used the RADTRAD-NAI computer code to model the time dependent transport of radionuclides, from the primary to secondary side and consequently to the environment via the ADVs. The licensee's analysis conforms with Appendix G of RG 1.183, which identifies acceptable radiological analysis assumptions for an LRA. The licensee assumed the same CR ventilation timing sequence as was used for the FHA.

The licensee evaluated the radiological consequences resulting from the postulated LRA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident dose criteria specified in SRP 15.0.1. The

NRC 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 9 and the licensee's calculated dose results are given in Table 1. The 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 LRA were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.1.6 Rod Control Cluster Assembly Ejection Accident

This accident is defined as the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of an RCCA and drive shaft. 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 and a limited amount of fuel melt are projected. The licensee assumed that as a result of localized fuel cladding damage, 10 percent of the gap activity is released to the primary coolant. In addition, the licensee assumes that 0.25 percent of the fuel inventory is also released to the primary coolant as a result of limited fuel melting. 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 (using design leakage assumptions)

• 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 10 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 0.01 for both halogens and noble gases. In addition, the licensee assumed that localized heating causes 0.25 percent of the fuel to melt, releasing 25 percent of the halogens and 100 percent of the noble gases contained in the melted fuel. As a result of the fuel melt portion of the fuel damage, the fraction of the core halogen activity released is 0.000625 (0.0025 x 0.25) and the fraction of noble gas activity released is 0.0025. The total activity released as a result of the fuel damage from the REA is the sum of the clad failure fraction and the fuel melt fraction.

DNC has determined that containment sprays will not initiate due to an REA 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 REA, 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 an SI signal will be initiated 2 minutes after the accident and, as a result, has assumed that the CR will not be isolated until 2 minutes 10 seconds following an REA.

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 core release fractions for halogens and noble gases are based on the assumed consequences of 10 percent failed fuel and 0.25 percent melted fuel, as in the containment release case. To evaluate the fuel clad failure portion of the fuel damage, the fraction of core activity released is 0.01 for both halogens and noble gases as in the containment release case. For the secondary release pathway, the licensee assumed that 50 percent of the iodines and 100 percent of the noble gases contained in the melted fuel are released to the RCS. Therefore, as a result of the fuel melt portion of the fuel damage the fraction of the core halogen activity released to the RCS is 0.00125 (0.0025 x 0.5) and the fraction of noble gas activity released is 0.0025.

For the secondary release case, the licensee assumed that fission products released from the fuel are instantaneously and homogeneously mixed in the RCS and transported to the secondary side of the SGs via primary-to-secondary leakage at the TS value of 1 gpm for 20 minutes. The licensee has determined that, for this event, a 20 minute time period is required for the primary system pressure to fall below the secondary side system pressure. A LOOP is conservatively assumed to occur at T = 0, 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 main steam safety valves. During the first 20 minutes of the accident, the only steam release is assumed to be via the secondary safety valve. When the primary system pressure drops below the secondary side pressure, the safety valve closes. At T = 2 hours, a cooldown to the RHRS entry conditions is initiated. Steam releases are assumed to begin again at T = 2 hours and continue until T = 18 hours, at which time shutdown cooling can be implemented to fully remove decay heat crediting only safety grade equipment. 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.

The licensee evaluated the radiological consequences resulting from the postulated REA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident dose criteria specified in SRP 15.0.1. The NRC 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 10 and the licensee's calculated dose results are given in Table 1. The 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 LRA were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.1.7 Control Room Habitability

The CREPS is designed to ensure that a positive pressure is maintained in the CR envelope for any event with the potential for radioactive releases. The positive pressure supplied by the CREPS limits CR inleakage and consequently dose to the CR occupants. With the implementation of the AST, none of the associated radiological analyses credit the CREPS in the calculation of the dose to the CR occupants. Since the acceptance criterion is met for the radiological DBAs without crediting the CREPS, the licensee has proposed to eliminate the TS associated with the system. However, the licensee is not physically removing the CREPS and has allowed time for the operation of the system, before initiating the CREVS, in the CR ventilation timing sequence for all the DBAs. The Staff's review of this TS change is in Section 3.1.9.3 of this SE.

The CR effective volume used in the habitability analyses is 2.38E+05 ft^3. The normal CR ventilation intake flow rate prior to isolation is 1,595 cfm. The CR will isolate on a control building isolation (CBI) signal from an SI signal for the LOCA and REA. For all other DBAs analyzed in this amendment request, the CR will isolate on a CBI signal from the CR inlet radiation monitor.

The period after CBI and prior to CREVS initiation is referred to as the CR neutral condition. During the neutral condition, the CR is isolated, the normal ventilation flow rate of 1595 cfm has terminated, and the CREVS is not operating. During the neutral condition there is no mechanically-induced ventilation of the CR. For the neutral condition, the licensee has conservatively assumed an unfiltered inleakage of 350 cfm for use in CR habitability calculations. The licensee assumes that the 350 cfm of unfiltered inleakage continues for an additional 40 minutes, after the time period allowed for CREPS operation (61 minutes), to enable the manual alignment of the CREVS. Therefore, the CR habitability analyses assume that the neutral condition described above, and the associated 350 cfm of unfiltered inleakage persists for a total of 101 minutes following CBI.

The licensee assumes that 101 minutes after CBI, the manual alignment of the CREVS will have been completed and the CREVS will be operational. The CREVS provides 230 cfm of filtered pressurization flow and 666 cfm of filtered recirculation flow to the CR. The period when the CREVS is operational is referred to as the CR positive pressure period. During the CR positive pressure period, the licensee assumes an unfiltered inleakage of 100 cfm, which is conservatively based on tracer gas testing results. Once initiated, CREVS operation and the associated CR positive pressure period persists for the duration of the event in the CR habitability analyses. The CREVS 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 CR ventilation assumptions are discussed in detail in the FHA portion of this SE in Section 3.1.2. The CR ventilation assumptions used in the FHA are also used in all other DBAs analyzed in this amendment request except the LOCA and the REA.

The LOCA causes a CBI signal to isolate the CR. The control building is isolated within 5 seconds after a CBI signal. Following the guidance of the AST, as described in RG 1.183, the onset of the gap release is not assumed to begin until 30 seconds after the initiation of the LOCA. Therefore, for the LOCA the licensee assumed that the CR will be isolated prior to the arrival of the radioactive release. The CR is assumed to be in a neutral condition for 101 minutes until the CREVS is operating.

For the REA, the CR ventilation assumptions are the same as the FHA with the exception that the CR isolation is initiated based on the SI signal which occurs at T = 2 minutes. Therefore, the CR is not isolated until 2 minutes and 10 seconds following an REA and the normal unfiltered ventilation flow of 1595 cfm persists for the first 2 minutes and 10 seconds of the REA.

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.

3.1.8 Atmospheric Dispersion Estimates

DNC used onsite meteorological data collected during calendar years 1997-2001 to generate new CR atmospheric dispersion factors (χ/Q values) for the MPS3, ground level releases associated with the dose assessment discussed above. The resulting χ/Q values represent a change from the χ/Q values used in the current MPS3 UFSAR, Chapter 15, "Accident Analyses." The licensee used previously-generated χ/Q values for postulated elevated releases from the MPS3 site stack to the CR. DNC also used previously-generated χ/Q values for both elevated and ground level releases to the EAB and LPZ.

3.1.8.1 Meteorological Data

DNC generated new CR χ/Q values for the MPS3 ground level release dose assessments using site meteorological data collected during 1997-2001. These data were provided for NRC staff review in the form of hourly meteorological data files (for input into the ARCON96 atmospheric dispersion computer code). 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 MPS3 site 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 and wind direction at both levels and stability (temperature difference) data was in the upper 90 percentiles during each of the 5 years. This meets the data recovery recommendation of RG 1.23, "Onsite Meteorological Programs." DNC stated that during 1997-2001, the onsite meteorological data were collected and processed in accordance with the guidance described in RG 1.23 and standards described in American National Standards Institute/American Nuclear Society (ANSI/ANS) 2.5-1984, "Standard for Determining Meteorological Information at Nuclear Power Sites."

With respect to the reported MPS3 site 1997-2001 atmospheric stability measurements, the duration of stable and unstable conditions were generally consistent with expected meteorological conditions. However, NRC staff noted a somewhat higher degree of year-to-year variability than expected in the frequency of occurrence of each stability category. Further, while stable and neutral conditions were generally reported to occur at night and unstable and neutral conditions during the day, with neutral or near-neutral conditions predominating during each year, there was a moderate occurrence of stable conditions during the day with lesser occurrence of unstable conditions at night. The NRC staff has judged that the net effect should result in a conservative estimate of the χ/Q values for this specific application because of the number of reported stable atmospheric conditions. Wind speed and wind direction frequency distributions for each measurement level were reasonably 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 MPS3 site provide an acceptable basis for making atmospheric dispersion estimates for use in the dose assessments performed in support of this specific license amendment request. However, these data should not be considered acceptable for use in other dose assessments or other meteorological applications without further review.

3.1.8.2 Control Room Atmospheric Dispersion Factors

MPS3 has a single CR air intake for both its normal and emergency ventilation. Releases resulting from the postulated LOCA and REA were modeled from the MPS3 site stack, turbine building vent, containment edge, main steam valve building roof vent, engineered safety features building roof vent, and RWST. The SG atmospheric dump and safety valves were assumed release points for the SGTR, LRA, and the secondary side of the REA. The turbine building roof vent was modeled as the release point associated with the MSLB. Releases from the MPS3 site stack were modeled as an elevated release. Releases from the containment 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. These new CR χ/Q values were calculated using the ARCON96 computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes"). RG 1.194 states that ARCON96 is an acceptable methodology for assessing CR χ/Q values for use in DBA 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 license amendment request for MPS3.

For the ground level releases, DNC used the shortest “taut string length” around or over intervening building structures to determine an “effective” horizontal distance between each release location and the CR intake. These taut string lengths factored in the difference in elevation between the source location and the CR intake. DNC then input the effective horizontal distance and the actual elevations of the source and intake to make the calculated slant path in the ARCON96 calculation match the shortest taut string length for each of the release/receptor pairs. DNC also addressed the most limiting cases for each pair including consideration of possible single failure and loss of offsite power. The resulting CR LOCA, FHA, SGTR, MSLB, LRA, and REA χ/Q values are listed in Table 2.

DNC used previously-generated CR χ/Q values for releases from the MPS3 site stack. These χ/Q values were approved as part of MPS3 License Amendment No. 211, dated September 16, 2002 (ADAMS Accession No. ML022470399). The χ/Q values were based upon the methodology described in RG 1.145, and include consideration of fumigation conditions. In the current license amendment request, DNC noted that the χ/Q values are conservative when compared to the options for determination of χ/Q values from elevated releases as recommended in RG 1.194 which was issued in June 2003, subsequent to Amendment No. 211. The NRC staff made limited comparative approximations and is in agreement with DNC’s conclusion.

In summary, the NRC staff qualitatively reviewed the inputs to the ARCON96 calculations and found them generally consistent with site configuration drawings and NRC staff practice. The NRC staff made an independent evaluation of the resulting atmospheric dispersion estimates by running the ARCON96 computer model for several random cases and obtained similar results. On the basis of the review described above and review of the SE associated with Amendment No. 211, the NRC staff has concluded that the χ/Q values for the MPS3 LOCA, FHA, SGTR, MSLB, LRA, and REA releases to the CR air intake presented in Table 2 are acceptable for use in the DBA CR dose assessment performed in support of this license amendment request.

3.1.8.3 Offsite Atmospheric Dispersion Factors

DNC used design-basis χ/Q values that were accepted by the NRC staff in License Amendment No. 211 to evaluate the impact of the MPS3 postulated LOCA, FHA, SGTR, MSLB, LRA, and REA radiological releases to the EAB and LPZ addressed in this amendment request. These χ/Q values were generated using the methodology described in RG 1.145. Based on the review described in the SE associated with Amendment No. 211 and a review of the licensee's use of these χ/Q values in the current license amendment request, the NRC staff has concluded that the EAB and LPZ χ/Q values presented in Table 3 are acceptable for use in the DBA assessments described in this SE.

3.1.9 TS Changes

3.1.9.1 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 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. To maintain consistency between the revised DBA analyses and the TSs, the licensee has proposed to revise the definition of DE I-131, in Section 1.10 of the TS definitions, to reference FGR 11 as the source of thyroid dose conversion factors deleting reference to RG 1.109, Revision 1. The following is the proposed revision:

> DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same CDE-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."

It should be noted that the licensee has proposed to modify the definition by making reference to the committed dose equivalent (CDE) thyroid dose. In an RAI (ADAMS Accession No. ML050040264), the NRC staff proposed that the TEDE dose be referenced in the definition of DE I-131 since the AST acceptance criteria is based on TEDE and no longer makes specific reference to the thyroid dose. The applicable DCFs for the calculation of the inhalation contribution to TEDE would be the CEDE DCFs. However, in this case, the numerical difference between using the DCFs for CDE thyroid vs CEDE for the calculation of DE I-131 is minimal (approximately 2 percent). Therefore, the NRC staff finds that the licensee retaining the reference to thyroid dose (CDE-thyroid) in the DE I-131 definition, and to use the CDE thyroid DCFs from FGR 11 is consistent with the guidelines for implementation of the AST and is acceptable.

3.1.9.2 TS 3/4.7.7, "Control Room Emergency Air Filtration System"

The licensee proposed to change TS 3/4.7.7, "Control Room Emergency Air Filtration System," Surveillance Requirements "c.2" and "d" to reflect a methyl penetration less than, or equal to, 5 percent for the CREAFs, instead of 2.5 percent. This proposed change was withdrawn by the licensee in a letter dated January 30, 2006.

3.1.9.3 TS 3/4.7.8 "Control Room Envelope Pressurization System"

The licensee proposed to delete TS 3/4.7.8, "Control Room Envelope Pressurization System." The CREPS is no longer credited in the accident analyses described in the evaluation.

The NRC staff reviewed the CREPS operation and requirements to determine if TS requirements could be deleted. The CREPS provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. The CREPS consists of two banks of air bottles, with its associated piping, instrumentation, and controls. Each bank is capable of providing the CR area with one hour of air following any event with the potential for radioactive releases.

Currently, during normal operations, the CREPS is required to be on standby. The CREPS is required to operate during post-accident operations to ensure that the CR will remain habitable during and following accident conditions.

In implementing the AST, the licensee did not take credit for the operation of the CREPS to reduce unfiltered in-leakage to the CR in the design-basis analyses. In the analyses, the licensee assumed an unfiltered inleakge of 350 cfm to the CR in the unpressurized condition. This number is conservative with respect to the results of tracer gas testing of the CR, which indicate an in-leakage of less than 100 cfm during a similar unpressurized condition.

TS 3/4.7.8 provides operability requirements, associated actions, and surveillance requirements for the CREPS. A review of 10 CFR 50.36(c)(2)(ii), which contains the requirements for items that must be in TSs, determined that the CR pressurization system does not meet the criteria that would require an LCO for the system. The technical requirements for the system will be relocated to other licensee controlled documents. As such, the NRC staff finds that the removal of TS requirements on the CR pressurization system is acceptable.

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

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 0.06 L_a. Therefore, to maintain consistency between the DBA radiological analyses and the TSs, the licensee proposed to change the leakage rate acceptance criteria for all penetrations that are secondary containment bypass leakage paths in Section 6.8.4.f, "Containment Leakage Testing Program," from # 0.042 L_a to # 0.06 L_a. Since the acceptance criterion is met for the AST implementation analyses assuming a leakage rate of 0.06 L_a for all penetrations that are secondary containment bypass leakage, the NRC staff finds this proposed change acceptable.

3.1.9.5 Bases Changes

The licensee stated that the Bases Sections 3/4.7.7 and 3/4.7.8 will be revised to reflect the above listed changes in accordance with the MPS3 Bases Control Program as described in Section 6.18 of the TSs.

The NRC staff considered the proposed changes as information only. The NRC staff did not review or make a finding with respect to these changes.

3.1.10 Summary

As described above, the NRC 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 MPS3. The NRC 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 NRC staff compared the doses estimated by the licensee to the applicable criteria identified in Section 2.0. The NRC staff also finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and CR doses will comply with these criteria. The NRC staff further finds reasonable assurance that MPS3, 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 accident source term in the DNC 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 RG1.183. 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 MPS3 design-basis, and modified by the present amendment.

The NRC staff finds that the proposed TS changes (excluding the proposed change which was withdrawn) are consistent with the Commission's regulations, that the proposed changes are acceptable, and that the proposed changes do not impact public health and safety and the safety of CR operators.

3.2 Containment Sump Water Chemistry

According to NUREG-1465, the iodine entering the containment of a reactor coolant system during an accident would be composed of at least 95 percent cesium iodide (CsI) (Reference 2). Upon deposition on interior surfaces and dissolution in the containment pool of a PWR, 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 3×10^{-4}. 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 MPS3, 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 amount of TSP required to maintain the pH above 7 for 30 days can be calculated using a combination of assumptions and known chemical relationships. The NRC staff reviewed the assumptions and methodology the licensee used to conclude that the pH in the pool would be maintained above 7 for the 30-day LOCA period. The NRC staff then performed manual calculations to evaluate the pool pH and the effect of backleakage on iodine concentration and pH in the RWST, which is another potential source of iodine release.

Hydroiodic acid (HI) is expected to be present in containments in small amounts (less than ~5 moles) and was not considered in the pH evaluations by the NRC staff or the licensee (Reference 3, page 3). Cesium hydroxide was also ignored in the pH evaluations, which adds conservatism since it is an alkaline species and therefore raises the pH (Reference 3, page 7).

3.2.1 Boric Acid from the RCS

The RCS, pressurizer, surge line, accumulators, and RWST contain borated water (2,900 parts per million boron) that has a nominal room-temperature (77 EF) pH of about 4.4. Hence, this is the initial pH in the pool when coolant is released at the onset of a LOCA.

3.2.2 Nitric Acid

HNO_3 is 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. By setting the 30-day dose rate equivalent to the one-year dose rate, the licensee calculated a conservative total of 1385 moles generated. This was based on a production rate of 7.3×10^{-6} moles HNO_3 per liter per MRad, which is consistent with NUREG/CR-5950.

3.2.3 Hydrochloric Acid

HCl is 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 used a conservative (one-year) dose rate and calculated a total of 7550 moles generated. This was based on a production rate of 4.6×10^{-4} moles HCl per pound of insulation per MRad, which is consistent with NUREG/CR-5950.

3.2.4 Trisodium Phosphate pH Buffer

The containment sump at MPS3 contains baskets of solid TSP ($Na_3PO_4 \cdot 12H_20 \cdot ¼NaOH$, or TSP), which dissolves as the water level in the sump rises during a LOCA. The TSP charge at MPS3 is 39" high, with the bottom located 6" above the containment mat elevation. The licensee assumed that the TSP would dissolve immediately and mix uniformly if it were contacted by rising water in the sump. The licensee also made the conservative assumption that the TSP would dissolve strictly in proportion to the level of the rising water. This means the TSP is not allowed, in the pH evaluation, to slump into the water as the base of the TSP dissolves. The final concentration of the 20,351 Kg of TSP in the pool would be 0.0133 mol/L. According to NUREG/CR-5950 (Reference 3, page 9), a TSP concentration of the order of 10^{-3} to 10^{-1} mol/L would maintain the pH level to within 0.1 unit at a pH near 7 when 10^3 moles of strong acid are added to a containment pool volume similar to that of MPS3.

3.2.5 Containment Pool pH Calculation

Using a different source than the licensee, the NRC staff determined the pH of mixtures of boric acid, TSP, HCl, and HNO_3 at various times during the 30-day LOCA period. The NRC staff's calculated pH values for these times are approximately the same as the licensee's. For example, the licensee reported a maximum pH of 8.1 at 50 minutes into the accident (when the TSP becomes fully immersed), followed by a gradual decrease to a final pH of about 7.1 starting at about 3.3 hours. The NRC staff's calculated pH values of 8.2 at 50 minutes and 7.1 at 30 days are consistent with those of the licensee. The NRC staff's calculated pH value of 7.1 at 30 days includes a decrease of 0.3 pH unit caused by the HCl and HNO_3 .

3.2.6 Back-leakage from the Sump to the RWST

The licensee stated that following a LOCA, valve realignment occurs to switch the ECCS from the RWST to the containment sump. This, the licensee stated, results in leakage through packing glands, pump shaft seals, and flanged connections. The licensee calculated the leakage flow rate for each source and a total volume of RWST back-leakage as a function of time. Based on the information provided by the licensee, 74,100 gallons of liquid containing 519 g of iodine would leak back to the RWST. When combined with the 47,655 gallons remaining in the RWST after the injection phase, the resulting iodine concentration in the RWST would be about 1×10^{-5} mol/L after 30 days. According to the NRC staff's calculations, the pH of the tank contents - including TSP, boron, and strong acids - would be in the range 6.9 - 7.1. At this concentration of iodine, no elemental iodine would be expected above pH ~5.5 (Reference 3, Figure 3.1). Although the pH is lower at earlier times, starting at about 4.4 when the tank contains no TSP, the iodine concentration is so low there is no significant conversion to elemental iodine (e.g., 10^{-9} mol/L I_2).

3.2.7 Summary

The NRC staff reviewed the licensee's assumptions, methodology, and conclusions regarding the variation of pH with time in the containment pool and RWST for the 30-day period following the LOCA. The NRC staff performed independent calculations to support the review. Based on these calculations, the NRC staff concluded that the pH of the containment pool would remain above 7 for 30 days and therefore prevent iodine release. The NRC staff also concluded iodine would not be released from the RWST as a result of back-leakage because of the low concentration of iodine when the tank pH is below 7, and the steady pH increase that would occur (due to TSP) as iodine concentration increases in the tank. Based on these results, the NRC staff finds the licensee's proposal acceptable.

3.3 <u>Core Fission Product Release</u>

The MPS3 equilibrium core inventory for radiological consequence calculations was determined from ORIGEN calculations. The NRC staff finds the use of ORIGEN acceptable in accordance with Section 3.1 of RG 1.183.

For the LOCA dose analysis, core release fractions are consistent with Table 2 of RG 1.183. The NRC staff finds these acceptable.

For the non-LOCA transients and accidents (including the FHA), the fuel/clad gap release fractions are consistent with RG 1.183. The NRC staff finds these acceptable. The NRC staff noted that the amendment did not request any exceptions to the criteria contained in RG 1.183. In particular, the amendment did not request any exceptions for fuel assemblies that would exceed the rod power/burnup applicability criteria in footnote 11 of the RG.

Based upon the above review, the NRC staff finds the determination of PWR core inventory and the use of the gap fractions from RG 1.183 for Non-LOCA transients and accidents acceptable for this amendment.

3.4 Technical Evaluation Conclusion

Based on the considerations discussed in SE Sections 3.1 through 3.3, the NRC staff finds the proposed amendment to be acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Connecticut State official was notified of the proposed issuance of the amendment. The State official agreed with the NRC staff"s conclusion as stated in Section 6.0 of this Safety Evaluation.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types of any effluents 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 (69 FR 55468). 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.

7.0 REFERENCES

1. RG 1.183, “Alternative Radiological Source Terms For Evaluating Design-Basis Accidents At Nuclear Power Reactors,” July 2000, NRC.

2. NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” NRC, Office of Nuclear Regulatory Research, February 1995.

3. NUREG/CR-5950, “Iodine Evolution and pH Control,” NRC Office of Nuclear Regulatory Research, December 1992.

4. USNRC Letter dated January 25, 1991, David H. Jaffe, (U.S. NRC) to E. J. Mroczka, (NNECo), “Millstone Nuclear Power Station, Unit No. 3 - Issuance of Amendment” (TAC No. 76066), ADAMS Accession No. ML011790140 (Amendment No. 59 to allow an increase in the normal containment pressure range).

5. MPS3 License Amendment Request Related to the Supplementary Leakage Collection and Release System (PLAR 3-98-5) Supplemental Information, December 20, 2001, ADAMS Accession No. ML020250052.

6. Amendment No. 211 to Facility Operating License No. NPF-49 for MPS3, September 16, 2002, ADAMS Accession No. ML022470399.

7. John A. Nakoski (NRC) to R.P. Necci (NNECo), “Millstone Nuclear Power Station, Unit No. 3 - Issuance of Amendment Re: Reactor Water Storage Tank Back Leakage” (TAC No. MA1749). Amendment No. 176,
ADAMS Accession No. ML993220168.

8. V. Nerses (NRC) to D. Christian, "Millstone Station Unit No. 3 - Issuance of Amendment Re: Selective Implementation of Alternative Source Term” (TAC No. MB8137), dated March 17, 2004, Amendment No. 219, ADAMS Accession No. ML040610926.

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Table 1
MPS3 Radiological Consequences Expressed as TEDE [1]
(rem)

Design-basis Accidents	EAB [2]	LPZ [3]	Control Room
Loss of Coolant Accident	9.1E+00	4.5E+00	3.4E+00
Dose Criteria	2.5E+01	2.5E+01	5.0E+00
Fuel Handling Accident	2.4E+00	1.3E!01	4.9E+00
Dose Criteria	6.3E+00	6.3E+00	5.0E+00
Steam generator tube rupture [4]	2.1E+00	1.8E!01	3.0E+00
Dose criteria	2.5E+01	2.5E+01	5.0E+00
Steam generator tube rupture [5]	9.0E!01	9.0E!02	1.3E+00
Dose criteria	2.5E+00	2.5E+00	5.0E+00
Main steamline break accident [4]	9.1E!02	3.6E!02	1.2E+00
Dose criteria	2.5E+01	2.5E+01	5.0E+00
Main steamline break accident [5]	3.6E!01	1.8E!01	3.0E+00
Dose criteria	2.5E+00	2.5E+00	5.0E+00
Locked rotor accident	2.3E+00	3.7E!01	3.2E+00
Dose criteria	2.5E+00	2.5E+00	5.0E+00
REA[6] Accident			
Containment	8.7E!01	4.8E!01	8.3E!01
Secondary side	1.2E!01	1.5E!02	5.3E!02
Dose criteria	6.3E+00	6.3E+00	5.0E+00

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

Table 2
Control Room Atmospheric Dispersion Factors

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

Table 2 (continued)
Control Room Atmospheric Dispersion Factors

Turbine Building		
	0 - 4 hours	5.40E-03
	4 - 8 hours	3.51E-03
	8 - 24 hours	1.38E-03
	24 - 96 hours	1.01E-03
	96 - 720 hours	8.49E-04

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

Receptor/ Source Location / Duration		χ/Q (sec/rn^3)
EAB (0 - 720 hours)		
Containment Release		5.42E-04
Millstone Site Stack Release (includes fumigation)		1.00E-04
Other Release Points		4.30E-04
LPZ		
Millstone Site Non-Stack Release Points		
	0 - 8 hours	2.91E-05
	8 - 24 hours	1.99E-05
	24 - 96 hours	8.66E-06
	96 - 720 hours	2.63E-06
Millstone Site Stack (includes fumigation)		
	0 - 4 hours	2.69E-05
	4 - 8 hours	1.07E-05
	8 - 24 hours	6.72E-06
	24 - 96 hours	2.46E-06
	96 - 720 hours	5.83E-07

Table 4
MPS3 Control Room Data and Assumptions

CR effective volume	2.38E+05 ft^3
Normal CR intake flow rate prior to isolation	1595 cfm
Unfiltered inleakage during periods of neutral pressure	350 cfm
Unfiltered inleakage during periods of positive pressure	100 cfm
CREVS recirculation flow rate	666 cfm
CREVS pressurization flow rate	230 cfm
Response time for CR inlet radiation monitor to generate CBI signal	5 seconds
Response time for CR to Isolate upon receipt of CBI	5 seconds
Time allotted for delay of CRE pressurization system (CREPS)	1 minute
Time allotted for CREPS discharge to the CR (CREPS is not credited in any dose analyses)	60 minutes
Time allotted for operator action to align CREVS after CREPS discharge	40 minutes
Total time allotted to place CREVS in service (summation of the 3 preceding time intervals)	101 minutes after CBI signal
Filter Efficiencies for CREVS	90% elemental 90% aerosol 70% organic
Containment wall thickness	4.5 ft concrete
Containment dome thickness	2.5 ft concrete
Control building wall thickness	2 ft concrete
CR ceiling thickness	8 inches concrete
Control building roof thickness	1 ft-10 in 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
MPS3 Data and Assumptions for the LOCA

Containment free air volume	2.35E+06 ft^3
Containment leak rate	0.3% weight% per day (L_a)
Containment bypass leak rate	$0.06L_a$
Containment leak rate reduction	50% after 24 hours (offsite analyses) 50% after 1 hour (CR analysis)
Secondary containment drawdown time	2 minutes
Iodine chemical form in containment atmosphere	95% cesium iodide 4.85% elemental iodine 0.15% organic iodine
Iodine chemical form in the sump and RWST	97% elemental 3% organic
Containment sump pH	$ 7
SLRCS filter efficiency	95% all iodines and particulates
Auxiliary building filter efficiency	95% all iodines and particulates
QSS effective operation period	72.5 - 7,480 seconds (0.02014 - 2.078 hours)
RSS start time	660 +/- 20 seconds
RSS effective time	14 minutes (T = 0.2333 hr)
Elemental iodine removal coefficient	20 per hour (0.02014 - 2.078 hrs)
QSS particulate iodine removal coefficient	DF < 50: 12.73 (0.02014 - 0.2333 hr)
Particulate iodine removal coefficient for QSS and RSS	DF < 50: 16.14 (0.2333 - 1.9 hrs) DF > 50: 1.61 (1.9 hrs - 2.078 hrs)
QSS containment coverage volume	1,166,200 ft^3
QSS and RSS containment coverage volume	1,515,858 ft^3
ECCS leakage outside containment	4,730 cc/hr
Minimum available RWST volume	1,072,886 gallons
Minimum QSS auto trip value	47,652 gallons
RWST maximum fill volume	1,206,644 gallons

Table 6
MPS3 Data and Assumptions for the FHA

Fuel clad damage	1 assembly plus 50 rods out of a total of 193 assemblies in a core
Gap fractions	
Noble gases	10%
Halogens	8%
Pool DF	
Noble gases	1
Iodines	200 (effective DF)
Release point	Turbine Building Ventilation Stack or Enclosure Building / Containment Ground
Decay time	100 hours
Radial peaking factor	1.7
Duration of release	2 hours
Control room ventilation timing:	
T= 0 seconds	Normal CR unfiltered intake flow -1595 cfm
T= 10 seconds	CR isolates on radiation monitor signal Intake flow - 0 cfm - neutral condition Assumed unfiltered inleakage - 350 cfm
T = 1.685 hours	CREVS filtered intake flow - 230 cfm Assumed unfiltered inleakage - 100 cfm CREVS filtered recirculation flow - 666 cfm

Table 7
MPS3 Data and Assumptions for the SGTR Accident

Primary-to-secondary leak rate TS limit	1 gpm (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
Credited operator actions (affected SG)	
Secure release from affected SG	2946 seconds
Credited operator actions (intact SGs)	
Secure release from intact SGs	18 hours
Chemical form of iodine released from SGs	Elemental 97% Organic 3%
Iodine PC	100
Moisture carryover in intact SGs	1%
Duration of release to environment	
Intact SGs	0 - 18 hours
Affected SG	0 - 0.8183 hours & 2 - 8 hours
Initial SG steam mass	8,870 lbm / SG
Initial SG liquid mass	97,222 lbm / SG
CR ventilation timing	Same as for the FHA

Additional assumptions:

Dose consequence from release of initial secondary side steam is not significant
CR plume and CR filter shine dose conservatively set at values from the LOCA analyses

Table 8
MPS3 Data and Assumptions for the MSLB Accident

Primary-to-secondary leak rate TS limit	1 gpm (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	Assumed to occur at accident initiation
Release points:	
Affected SG	Turbine building
Intact SG	ADVs
Iodine PC for intact SGs	100
Moisture carryover in intact SGs	1%
Primary-to-secondary leakage	
Affected SG	500 gpd
Total	1 gpm
SG liquid mass	164,200 lbm
CR ventilation timing	Same as FHA
Duration of SG release:	
Affected SG	55.2 hours
Intact SG	18 hours
Steam release from affected SG	
Initial inventory	1.75E+05 lbm/min (0 - 56.3 sec)
Primary-to-secondary leak	2.918 lbm/min (0 - 55.2 hrs)
Steam Release from intact SGs	
0 - 2 hours	3.41E+03 lbm/min
2 - 8 hours	2.73E+03 lbm/min
8 - 18 hours	4.56E+03 lbm/min

Additional assumptions:

MPS3 auxiliary feed system is available to maintain water level in intact SGs
CR plume and CR filter shine dose conservatively set at values from the LOCA analyses

Table 9
MPS3 Data and Assumptions for the LRA Accident

Fuel clad failure	7%
Radial peaking factor	1.7
Primary-to-secondary leak rate	0.35 gpm (affected SG) 0.65 gpm (intact SGs)
Release points	SG ADV
Licensing basis credited operator actions	Closure of ADV after 20 minutes
Chemical form of iodine released from the SGs to the environment	3% organic iodide 97% elemental iodine
Fraction of fission product inventory in gap	
Halogens	0.8
Noble gases	0.10
Alkali metals	0.12
Iodine PC in intact SG	100
Intact SG tube uncovery	None
Affected SG tube uncovery	Immediate dry out; 100% flashing assumed
Discharge rate from the affected SG ADV	820,000 lb/hr
Release Duration	
Intact SGs	18 hours
Affected SG	20 minutes
Total mass of steam to atmosphere from intact SGs	
0 - 2 hours	251,000 lbm
2 - 8 hours	1,031,000 lbm
8 - 11 hours	820,800 lbm
11-18 hours	1,915,359 lbm
Mass flow rates from intact SGs	
0 - 2 hours	1.255E+05 lbm/hr
2 - 8 hours	1.718E+05 lbm/hr
8 - 11 hours	2.736E+05 lbm/hr
11 - 18 hours	2.736E+05 lbm/hr

Table 9 (continued)
MPS3 Data and Assumptions for the LRA Accident

Moisture carryover in intact SG	1%
Initial SG liquid mass	4.414E+07 grams per SG
MPS3 SG ADV maximum flow rate	820,000 lbm/hr
CR ventilation timing	Same as the FHA

Additional assumptions:

Dose consequences are from release of initial secondary side liquid
Dose consequences from the release of steam is not significant
CR plume and CR filter shine dose conservatively set at values from the LOCA analyses

Table 10
MPS3 Data and Assumptions for the REA

Containment free air volume	2.35E+06 ft^3
Fraction of fuel clad failure	0.1
Fraction of core inventory in gap	
Noble gasses	0.1
Iodine	0.1
Fraction of core fuel melt	0.0025
Release fractions for melted fuel	
Containment release	
Noble gasses	1.0
Iodines	0.25
Reactor coolant release	
Noble gasses	1.0
Iodines	0.5
SI signal initiated after REA	2 minutes
Chemical form of iodine released from the SG to the environment	3% organic iodide 97% elemental iodine
Total primary-to-secondary leakage through all SGs	1 gpm
Time for primary system pressure to fall below secondary system pressure	1,200 seconds
Duration of steam releases	18 hours
Steam released from 0 to 1,200 seconds (primary system depressurization)	200,000 lbm
Steam released from 2 - 11 hours	1.547E+06 lbm
Steam released from 11 - 18 hours	1.916E+06 lbm
CR ventilation timing	
Time for CR isolation	2 minutes 10 seconds
Remainder of CR ventilation timing	Same intervals after CBI as the FHA

Millstone Power Station, Unit No. 3

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