September 20, 2004

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

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: SELECTIVE IMPLEMENTATION OF ALTERNATE SOURCE TERM (TAC NO. MB6479)

Dear Mr. Christian:

The Commission has issued the enclosed Amendment No. 284 to Facility Operating License No. DPR-65 for Millstone Power Station, Unit No. 2, in response to your application dated September 26, 2002, as supplemented June 2, 2003, May 7, June 18, and August 24, 2004.

The amendment proposes Technical Specification (TS) changes requested by Dominion Nuclear Connecticut, Inc. for Millstone Power Station, Unit No. 2. The proposed TS changes are based on a re-analysis of fuel-handling accidents. The revised analysis of these accidents is based on selective implementation of alternate source term methodology.

A copy of the related 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, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures: 1. Amendment No. 284 to DPR-65
2. Safety Evaluation

cc w/encls: See next page

cc:

Lillian M. Cuoco, Esquire
Senior Counsel
Dominion Resources Services, Inc.
Rope Ferry Road
Waterford, CT 06385

Edward L. Wilds, Jr., Ph.D.
Director, Division of Radiation
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

First Selectmen
Town of Waterford
15 Rope Ferry Road
Waterford, CT 06385

Charles Brinkman, Director
Washington Operations Nuclear Services
Westinghouse Electric Company
12300 Twinbrook Pkwy, Suite 330
Rockville, MD 20852

Senior Resident Inspector
Millstone Power Station
c/o U.S. Nuclear Regulatory Commission
P.O. Box 513
Niantic, CT 06357

Mr. W. R. Matthews
Senior Vice President - Nuclear Operations
Dominion Nuclear Connecticut, Inc.
Rope Ferry Road
Waterford, CT 06385

Mr. P. J. Parulis
Manager - Nuclear Oversight
Dominion Nuclear Connecticut, Inc.
Rope Ferry Road
Waterford, CT 06385

Mr. J. Alan Price
Site Vice President
Dominion Nuclear Connecticut, Inc.
Rope Ferry Road
Waterford, CT 06385

Mr. John Markowicz
Co-Chair
Nuclear Energy Advisory Council
9 Susan Terrace
Waterford, CT 06385

Mr. Evan W. Woollacott
Co-Chair
Nuclear Energy Advisory Council
128 Terry's Plain Road
Simsbury, CT 06070

Ms. Nancy Burton
147 Cross Highway
Redding Ridge, CT 00870

Mr. G. D. Hicks
Director - Nuclear Station Safety and Licensing
Dominion Nuclear Connecticut, Inc.
Rope Ferry Road
Waterford, CT 06385

Mr. S. E. Scace
Assistant to the Site Vice President
Dominion Nuclear Connecticut, Inc.
Rope Ferry Road
Waterford, CT 06385

Mr. Chris L. Funderburk
Director, Nuclear Licensing and
Operations Support
Dominion Resources Services, Inc.
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

cc:

Mr. A. J. Jordan, Jr.
Director - Nuclear Engineering
Dominion Nuclear Connecticut, Inc.
Rope Ferry Road
Waterford, CT 06385

Mr. S. P. Sarver
Director - Nuclear Station Operations
and Maintenance
Dominion Nuclear Connecticut, Inc.
Rope Ferry Road
Waterford, CT 06385

Mr. David W. Dodson
Licensing Supervisor
Dominion Nuclear Connecticut, Inc.
Rope Ferry Road
Waterford, CT 06385

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Sr. Vice President and Chief Nuclear Officer
Dominion Nuclear Connecticut, Inc.
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/RA/

Victor Nerses, Senior Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures: 1. Amendment No. 284 to DPR-65
2. Safety Evaluation

DISTRIBUTION:

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Package: ML

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DATE	9/15/04	9/15/04	9/2/04	SEs dated June 30, and August 27, 2004	9/7/04	9/20/04

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<u>DOMINION NUCLEAR CONNECTICUT, INC., ET AL.</u>

<u>DOCKET NO. 50-336</u>

<u>MILLSTONE POWER STATION, UNIT NO. 2</u>

<u>AMENDMENT TO FACILITY OPERATING LICENSE</u>

Amendment No. 284
License No. DPR-65

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

    A. The application for amendment by the applicant dated September 26, 2002, as supplemented June 2, 2003, May 7, June 18, and August 24, 2004, 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 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;

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 284, 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 the date of issuance, and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Daniel S. Collins, Acting Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 20, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 284

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

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

Remove	Insert
IX	IX
3/4 3-24	3/4 3-24
3/4 3-25	3/4 3-25
3/4 3-26	3/4 3-26
3/4 3-27	3/4 3-27
3/4 3-36	3/4 3-36
3/4 7-16	3/4 7-16
3/4 7-16a	3/4 7-16a
3/4 9-4	3/4 9-4
3/4 9-5	3/4 9-5
3/4 9-8	3/4 9-8
3/4 9-8a	3/4 9-8a
3/4 9-8b	3/4 9-8b
3/4 9-16	3/4 9-16
3/4 9-17	3/4 9-17
3/4 9-18	3/4 9-18
B 3/4 3-2a	B 3/4 3-2a
B 3/4 3-6	B 3/4 3-6
B 3/4 7-4a	B 3/4 7-4a
B 3/4 7-4b	B 3/4 7-4b
B 3/4 7-4c	B 3/4 7-4c
B 3/4 9-1a	B 3/4 9-1a
-------------	B 3/4 9-1b
B 3/4 9-2a	B 3/4 9-2a
B 3/4 9-3	B 3/4 9-3
B 3/4 9-3a	B 3/4 9-3a
B 3/4 9-3b	B 3/4 9-3b

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 284

TO FACILITY OPERATING LICENSE NO. DPR-65

DOMINION NUCLEAR CONNECTICUT, INC.

MILLSTONE POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

1.0 INTRODUCTION

By letter dated September 26, 2002 (ML023040334), as supplemented by letters dated June 2, 2003 (ML031610928), May 7 (ML041320350), June 18 (ML041740354), and August 24, 2004 (ML042390046), Dominion Nuclear Connecticut, Inc. (DNC or the licensee), requested a license amendment for Millstone Power Station, Unit No. 2 (MP2). The amendment will allow the current accident source term used in selected design-basis accident (DBA) radiological analyses to be replaced with an alternative source term (AST) pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67, “Accident Source Term.” This is a selective implementation of the AST as defined in Regulatory Guide (RG) 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.” DNC also proposed revisions to several technical specifications (TSs) related to containment (CNMT) and fuel building exhaust filter system operation during refueling periods. Conforming changes will also be made to the TS bases.

The June 2, 2003, May 7, June 18, and August 24, 2004, letters provided clarifying information that did not change the scope of the initial application as described in the *Federal Register* notice dated November 2, 2002 (67 FR 68731), and did not change the Nuclear Regulatory Commission (NRC or the Commission) staff’s initial proposed no significant hazards consideration determination.

The proposed changes requested by this application would revise the TSs to allow relaxation of containment operability requirements while handling irradiated fuel and core alterations. Specifically, the proposed changes would revise:

(1) TS 3.3.3.1, “Monitoring Instrumentation, Radiation Monitoring”

(2) TS 3.3.4, “Instrumentation, Containment Purge Valve Isolation Signal”

Relocate Surveillance Requirement (SR) 4.3.3.1.2 to TS 3.3.4 as SR 4.3.4.2. Modify the new SR 4.3.4.2 to require verification of the trip value. Revise the applicability of TS 3.3.4 to delete applicability “during CORE ALTERATIONS with the CNMT purge valves open” and “during the movement of irradiated fuel assemblies inside CNMT with the CNMT purge valves open," and replace these with "MODES 1, 2, 3, and 4." Action statements of TS 3.3.4 will be changed to be consistent with the revised applicability.

Revise TS Table 3.3-6: change applicability of Items 2.a. and 2.b. from "ALL MODES" to "1, 2, 3, & 4"; change Action 14; delete note "**"; delete Item 1.a; and delete Item 1.a, note "*." Revise TS Table 4.3-3 to delete Item 1.a; delete note "*"; and change Items 2.a and 2.b to be applicable only in Modes 1, 2, 3, and 4.

The effect of these proposed changes is to allow plant operation without the capability for an automatic purge valve closure or automatic diversion of spent fuel storage area ventilation during core alterations and movement of irradiated fuel.

(3) <u>TS 3.7.6.1, "Plant Systems, Control Room Emergency Ventilation System"</u>

Revise this TS to limit applicability to Modes 1 through 4 and during movement of irradiated fuel within the CNMT or spent fuel pool. DNC modified this request in its letter of May 7, 2004, to restore operability requirements during Modes 5 and 6. The effect of this proposed change, as revised, is to allow plant operation without the capability for control room isolation and recirculation filtration during movement of new fuel and movement of a shielded cask over the spent fuel pool cask laydown area.

(4) <u>TS 3.9.4, "Refueling Operations, Containment Building Penetrations"</u>

Revise limiting condition for operation (LCO) Item a to replace the existing language with the phrase, "The equipment door shall be either (1) closed and held in place by a minimum of four bolts, or (2) open under administrative control and capable of being closed and held in place by a minimum of four bolts." A new note "*" defines the required administrative controls. Revise LCO Item b.2 to delete the phrase "with containment purge in operation," and make reference to the new note "*." Revise LCO Item c.2 to replace "by an OPERABLE Containment Purge Valve Isolation System" with "under administrative control." Revise the applicability statement to remove " during CORE ALTERATIONS." Delete SR 4.9.4.2. The effect of these proposed changes is to allow movement of irradiated fuel inside CNMT with the equipment door open and without the operability of automatic isolation of CNMT purges.

(5) <u>TS 3.9.8.1, "Refueling Operations, Shutdown Cooling and Coolant Circulation - High Water Level"</u>

Revise TS 3.9.8.1 to delete Note 3.c.3).b) and Action c.3.b. The effect of this change is to eliminate the CNMT purge valve isolation system as a required means of isolating CNMT penetrations. Without an automatic isolation, the CNMT purge valves would be required to be shut by Note 3.c.3).a) and Action c.3.a.

(6) <u>TS 3.9.8.2, "Refueling Operations, Shutdown Cooling and Coolant Circulation - Low Water Level"</u>

Revise TS 3.9.8.2 to delete Action b.3.c).2). The effect of this change is to eliminate the CNMT purge valve isolation system as a required means of isolating CNMT penetrations. Without an automatic isolation, the CNMT purge valves would be required to be shut by Action b.3.c).1).

(7) TS 3.9.15, "Refueling Operations, Storage Pool Area Ventilation System"

Delete TS 3.9.15. The effect of this proposed change is to allow movement of irradiated fuel in the spent fuel pool area without satisfying current requirements for spent fuel area integrity, storage pool area ventilation system, and operability of an enclosure building filtration train.

2.0 REGULATORY EVALUATION

The construction permit for MP2 was issued by the Atomic Energy Commission (AEC) on December 11, 1970. The plant was designed and constructed based on the proposed General Design Criteria published by the AEC in the *Federal Register* (32 FR 10213) on July 11, 1967 (hereinafter referred to as "proposed GDC"). The AEC published the final rule that added Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "General Design Criteria for Nuclear Power Plants," in the *Federal Register* (36 FR 3255) on February 20, 1971 (hereinafter referred to as "GDC").

Differences between the proposed GDC and the GDC included a consolidation from 70 to 64 criteria. As discussed in the NRC's Staff Requirements Memorandum for SECY-92-223, dated September 18, 1992 (Agencywide Document Access and Management System (ADAMS) Accession No. ML003763736), the Commission decided not to apply the GDC to plants with construction permits issued prior to May 21, 1971. At the time of promulgation of Appendix A to 10 CFR Part 50, the Commission stressed that the GDC were not new requirements and that the GDC were promulgated to more clearly articulate the licensing requirements and practice in effect at that time. Each plant licensed before the GDC were formally adopted was evaluated on a plant-specific basis, determined to be safe, and licensed by the Commission. Because MP2's construction permit was issued prior to May 21, 1971, the requirements applicable to MP2 are those of the proposed GDC.

Applicants for license amendments are required by 10 CFR 50.91 to provide an analysis of significant hazards considerations, including increases in the consequences of accidents previously evaluated. These evaluations are performed to demonstrate that, in the event of an accident, radiation doses to persons onsite and offsite will continue to meet applicable acceptance criteria. Regulatory guidance for these evaluations is provided in the form of RGs and standard review plans. Fundamental to these evaluations is the source term -- the assumptions related to the radioactive material available for release to the environment. DBA analyses have traditionally used the source term provided in the 1962 document "Calculation of Distance Factors for Power and Test Reactor Sites," 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 NRC and the nuclear industry after the accident at Three Mile Island (TMI). In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," which 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 issued a new regulation, 10 CFR 50.67, "Accident Source Term," which provided a mechanism for licensed power reactors to replace the traditional accident source term used in their DBA analyses with an AST.

The NRC staff also issued regulatory guidance in using the AST in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

A licensee seeking to use an AST is required, pursuant to 10 CFR 50.67, to apply for a license amendment. An evaluation of the consequences of affected DBAs is required to be included with the submittal. DNC's application of September 26, 2002, as supplemented by letters dated June 2, 2003, May 7 and June 18, 2004, addresses these 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 MP2. As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) will replace the previous whole-body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, GDC-19 as the MP2 licensing basis with regard to the radiological consequences of the design-basis fuel-handling accidents (FHAs).

The NRC staff's safety evaluation (SE) provided below addresses the impact of the proposed changes on previously analyzed design-basis radiological consequences and the acceptability of the revised analysis results. The regulatory requirements for which the staff based its acceptance are the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 of RG 1.183 and GDC-19. Except where the licensee proposed a suitable alternative, the staff utilized the guidance in RG 1.183 in performing this review. The staff also considered relevant information in the MP2 Updated Final Safety Analysis Report (UFSAR). Other regulatory requirements for which the NRC staff based its acceptance are as follows:

1. Proposed GDC 17 (similar to GDC 64) - *Monitoring Radioactivity Releases*. Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and accident conditions.

2. Proposed GDC 18 (similar to GDC 63) - *Monitoring Fuel and Waste Storage*. Monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

3. Proposed GDC 69 (similar, in part, to GDC 61) - *Protection Against Radioactivity Release From Spent Fuel and Waste Storage*. Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

4. Proposed GDC 70 (similar, in part, to GDC 61) - *Control of Release of Radioactivity to the Environment*. The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous , liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10 CFR 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of

> occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

The NRC staff's evaluation of the acceptability of some of the proposed TS changes is based upon 10 CFR 50.36, "Technical Specifications." Section 50.36(c)(2)(ii) of 10 CFR requires that a TS LCO of a nuclear reactor must be established for each item meeting one or more of the following criteria:

Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

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

A licensee seeking to delete a functional unit from the TS LCO must demonstrate that these Criterion no longer apply to the functional unit to be deleted.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the technical analyses related to the radiological consequences of design-basis FHAs at MP2 that were performed by DNC in support of this proposed license amendment. Information regarding these analyses was provided by the licensee in Attachment 1 of the September 26, 2002, submittal, as supplemented by the June 2, 2003, May 7, June 18, and August 24, 2004, letters. The NRC staff reviewed the assumptions, inputs, and methods used by DNC in their analyses. The staff performed independent calculations to confirm the acceptability of the DNC analyses. However, the findings of this SE are based on the descriptions of the DNC analyses and other supporting information docketed by DNC. Only docketed information was relied upon in making this safety finding. DNC determined, and the staff concurs, that the proposed changes have a potential effect on the three previously analyzed DBAs: a FHA within the CNMT, a FHA in the fuel pool area and a spent fuel cask drop accident. DNC determined the TEDE at the exclusion area boundary (EAB) for the worst 2-hour period and the 0-30 day low population zone (LPZ) TEDE. DNC also evaluated the potential TEDE to control room personnel from these events.

3.1 FHA Radiological Consequence Analysis

This accident analysis postulates that a spent fuel assembly is dropped during refueling. The affected assembly is assumed to be the assembly with the highest inventory of fission products of the 217 assemblies in the core. All of the fuel rods in the assembly are conservatively assumed to rupture, releasing the radionuclides within the fuel rod to the fuel pool or reactor cavity water. Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod clad. 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 depending on their physical and chemical form. DNC assumed no decontamination for noble gases, a decontamination factor of 200 for radioiodines, and retention of all aerosol and particulate fission products. DNC assumed that 100 percent of the fission products released from the reactor cavity or spent fuel pool are released to the environment in 2 hours without any credit for filtration, holdup, or dilution. Since the revised assumptions and inputs are identical for the FHA within CNMT and the FHA outside CNMT, the results of the two events are identical. The revised analysis of the FHA outside CNMT no longer distinguishes between fuel that has decayed for more than 60 days or which has decayed for less than 60 days.

The assumptions found acceptable to the NRC staff are presented in Table 1. The EAB, LPZ, and control room doses estimated by DNC for the FHAs were found to be acceptable. Control room unfiltered inleakage is periodically assessed using tracer gas methods as required by existing TSs. The NRC staff performed independent calculations and confirmed the DNC conclusions.

3.2 Spent Fuel Cask Drop Accident Radiological Consequence Analysis

This accident analysis postulates that a spent fuel cask is dropped during refueling striking and damaging 1560 fuel assemblies. Of these, 184 assemblies are assumed to have decayed for 1 year and the remainder for 5 years. Administrative controls limit the age of the fuel assemblies in the area of potential impact. All of the fuel rods in the damaged assemblies are assumed to rupture, releasing the radionuclides within the fuel rod to the fuel pool. Due to the extended decay period, the dose significant radionuclides are the long-lived radionuclides Kr-85 and I-129. These fission products are decontaminated by passage through the pool water, depending on their physical and chemical form. DNC assumed no decontamination for noble gases, a decontamination factor of 200 for radioiodines, and retention of all aerosol and particulate fission products. DNC assumed that 100 percent of the fission products released from the spent fuel pool are released to the environment in 2 hours without any credit for filtration, holdup, or dilution.

The assumptions found acceptable to the NRC staff are presented in Table 1. The EAB, LPZ, and control room doses estimated by DNC for the FHAs were found to be acceptable. Control room unfiltered inleakage is periodically assessed using tracer gas methods as required by existing TSs. The staff performed independent calculations and confirmed the DNC conclusions.

3.3 Alternative Gap Fractions

By letter dated May 7, 2004, DNC proposed an alternative method to determine bounding gap fractions. This method was proposed in response to Footnote 11 to Table 3 of RG 1.183. Subsequent to their September 26, 2002, submittal, DNC determined that the linear heat generation rate (LHGR) limitation in the footnote might not be met for a small number of rodsand decided to utilize the alternative method suggested in the footnote. However, the NRC staff found that the DNC-proposed alternative method lacked sufficient detail for the staff to determine its acceptability.

By letter dated August 24, 2004, DNC proposed another alternate method for developing the FHA gap fractions used in the AST analysis for a fuel assembly that exceeds the RG 1.183 criteria of a peak rod average burnup greater than 54 GWD and a LHGR greater than 6.3 kw/ft. DNC proposed to use the gap fractions provided in RG 1.25, "Assumptions Used for Evaluating the Potential Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," with the modifications proposed in NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Reactors." NUREG/CR-5009 modified the I-131 contributions by 20 percent to account for the increased fission gases created by the increased burnup. DNC has indicated that MP2 will use these gap fraction values for 100 percent of the rods in the assembly when analyzing the source term even if less than 100 percent of the rods exceed the RG 1.183 criteria. The NRC staff finds this method acceptable because the proposed gap fractions are consistent with the current staff guidance on gap fractions for extended burnup.

The gap fractions proposed by DNC are tabulated in Figure 1 along with the RG 1.183 Table 3 values.

Figure 1 Gap Fractions

Nuclide Group	RG 1.183 Table 3 Gap Fractions	Proposed Gap Fraction
I-131	0.08	0.12
Kr-85	0.10	0.30
Other Noble Gas	0.05	0.10
Other Halogens	0.05	0.10
Alkali Metals	0.12	n/a

In the August 24, 2004, letter DNC dispositioned the impact of the increases in the gap fractions on the analyses described in the original submittal. DNC noted that since particulates are retained in the pool, any increase in alkali metal gap fractions could not impact the previous analyses. For the cask drop accident, DNC re-calculated the doses and determined that the increased gap fractions would increase the previously estimated doses, but that the doses remained a small fraction of the acceptance criteria. The NRC staff used the updated gap fractions in its independent calculations and confirmed DNC's conclusion.

3.4 Technical Specifications

3.4.1 TS 3.3.3.1 Monitoring Instrumentation, Radiation Monitoring

3.4.1.1 The licensee proposes to delete Item 1.a, “Area Monitors - Spent Fuel Storage and Ventilation System Isolation.” Note “*”, and Action 13 from Table 3.3-6 will also be deleted.

These monitors provide a signal to direct ventilation exhaust from the spent fuel storage area through a filter train. The licensee has shown on the basis of the FHA DBA that the fuel building exhaust filter system is not required to satisfy the dose guidelines of 10 CFR 50.67. Thus, the system is not on the primary success path for a DBA. As such, Criterion 3 of 10 CFR 50.36 does not require an LCO to be placed in the TS. The Action Item 13 is also not required. Note “*” which specifies applicability to fuel being in the storage building is also not required with the deletion of this section.

The NRC staff’s finding of acceptability for this item is only associated with the removal of the item from the TSs. Prior to removing any equipment or changing any procedure affecting the operation of engineering safeguards equipment, the licensee must use the appropriate modification process (10 CFR 50.59 or 10 CFR 50.90) to assure that the facility complies with all other commitments including GDC (or corresponding proposed GDC) as stated in 10 CFR Part 50, Appendix A or their equivalents, that the appropriate changes are made to the facility UFSAR and that defense-in-depth and safety margins are adequate. In particular, proposed GDC 18 (similar to GDC 63) requires appropriate systems “to detect conditions that may result in... excessive radiation levels”. Although the automatic isolation signal is not required, the alarm function may be needed to satisfy proposed GDC 18 and alert operators to the need to implement the procedural guidance which would isolate the spent fuel building.

The licensee states that “procedural guidance will be available for closing fuel building area atmosphere boundary penetrations if a[n] FHA occurs inside the fuel building.” The use of this procedural guidance will be implemented “as a defense in depth measure to minimize actual releases to the outside atmosphere much lower than assumed in the AST FHA analyses dose calculations.” The NRC staff concurs that the development and implementation of procedural guidance will increase defense in depth and facilitate managing releases during an FHA. As such it will provide increased protection to public health and safety.

3.4.1.2 The licensee proposes to change the applicability of Item 2.a, “Containment Atmosphere- Particulate” and Item 2.b, “Containment Atmosphere- Gaseous” so that they are no longer required in Modes 5 and 6 but are still required in Modes 1-4. The alarm/setpoint will be marked “n/a.” Action Item 14 is being changed to reference the actions required in TS Section 3.4.6.1. Note “**” is being deleted.

These process monitors currently have several functions. While refueling, they provide a Containment purge valve isolation signal on increased airborne radioactivity levels in Containment. They also aid in the detection of a leak in the reactor coolant system by monitoring airborne radioactivity levels in Containment. During refueling, the monitors provide one means of compliance with proposed GDC 17 (similar to GDC 64) which requires monitoring of releases from the containment in normal and accident conditions.

The licensee states that "the FHA Analyses do not assume automatic closure of the Containment purge valves on increasing airborne radioactivity levels." As a consequence, the automatic isolation of the containment purge system is no longer on the primary success path of a DBA. Thus, the requirement for an LCO based on Criterion 3 of 10 CFR 50.36 does not apply during refueling Modes 5 and 6. As a consequence, changing the applicability to modes 1, 2, 3 and 4 and marking the alarm/setpoint not applicable (n/a) is acceptable.

Operability requirements and appropriate actions for the Containment Atmosphere Particulate and Gaseous Process monitors for their role in leakage detection during Modes 1, 2, 3, and 4 are specified in TS 3.4.6. The actions specified in TS 3.4.6.1 are adequate for the LCO and it is appropriate for Action 14 in Table 3.6 to reference these actions. The NRC staff finds the proposed change to Action 14 acceptable.

Removal of automatic isolation of the purge system does reduce defense-in-depth. The licensee compensates for the reduction in defense-in-depth by adding a footnote to TS 3.9.4 discussed below which requires administrative controls to close the containment including isolation of the containment purge in 30 minutes with designated personnel after an FHA. The staff has determined that these administrative controls provide an important element of defense-in-depth and assures that the licensee will manage the consequences of an FHA in a manner that will afford adequate protection to the public. As such, the NRC staff finds that the removal of automatic isolation is acceptable with the addition of administrative controls to achieve closure and isolation.

Although the staff finds the removal of automatic isolation of the purge system from the TSs to be acceptable, the staff finds that the licensee must still meet the requirements of proposed GDC 17 (similar to GDC 64) for monitoring releases. An LCO for these requirements does not have to be in the TSs since it is not directly related to a DBA; however, the licensee should address monitoring of releases during refueling in a controlled document such as the technical requirements manual. The existing Action Item 14 had provisions for a constant air monitor or grab samples to be used in the event that the process monitors were not operational. Although this provision may be removed from the TSs, the NRC staff finds that the licensee must still show compliance with proposed GDC 17 which may be based on the function of these process monitors, a constant air monitor or grab samples.

The staff finds that Note "**" can be deleted since Type A integrated leak rate testing is only performed during Mode 5 and operability is being changed to Modes 1, 2, 3, and 4.

3.4.1.3 The licensee proposes to relocate SR 4.3.3.1.2 to 3/4.3.4 Containment Purge Valve Isolation Signal by incorporating it into SR 4.3.4.2. SR 4.3.3.1.2 will be marked deleted.

SR 4.3.3.1.2 provides the criteria for the determination and verification of the trip value for the Containment purge valve isolation signal from the containment atmosphere gaseous and particulate monitors. The relocation of this SR is acceptable as it is consistent with the relocation of the Containment purge valve isolation signal function of Containment atmosphere particulate and gaseous monitors from TS 3.3.3.1 to TS 3.3.4.

The effect of the proposed changes to these TSs is to allow plant operation without the capability for an automatic diversion of spent fuel storage area ventilation during core alterations and movement of irradiated fuel. In performing the re-analyses of the FHAs and spent fuel cask drop accidents, DNC did not credit any filtration, holdup, or dilution prior to release to the environment during a DBA FHA. The events were conservatively modeled as if there was no enclosing structure (CNMT or spent fuel building). Since the operability of the design feature is no longer assumed as an initial condition in a DBA analysis, the requested changes are acceptable with regard to DBA radiological consequences.

3.4.2 TS 3.3.4 Instrumentation, Containment Purge Valve Isolation Signal

3.4.2.1 The licensee proposes to change the Applicability from

> "During CORE ALTERATIONS with the containment purge valves open.
>
> During the movement of irradiated fuel assemblies inside containment with the containment purge valves open."
>
> to
>
> "MODES 1, 2, 3, and 4"

The licensee states that "the FHA Analyses do not assume automatic closure of the "Containment purge valves on increasing airborne radioactivity levels." As a consequence, the automatic isolation of the containment purge system is no longer on the primary success path of a DBA. Thus, the requirement for an LCO based on Criterion 3 of 10 CFR 50.36 does not apply during refueling Modes 5 and 6. The NRC staff concurs that the applicability statements "During CORE ALTERATIONS with the containment purge valves open" and "During the movement of irradiated fuel assemblies inside containment with containment purge valves open." can be deleted. Although the revised FHA analysis does not credit closure, the licensee stated that administrative controls will be developed for manual closure of the containment purge valves within 30 minutes of an FHA as a defense-in-depth measure.

The addition of applicability to Modes 1, 2, 3 and 4 is conservative. Although TS 3.6.3.2, "Containment Ventilation System" requires that the containment purge valves be closed in Modes 1, 2, 3, and 4, an isolation signal will be provided in these modes in the event that the valves might be open. The NRC staff finds this to be acceptable.

3.4.2.2 The licensee proposes to replace the Action statements a., b., and c. in TS 3.3.4 with:

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

The proposed change in the Action statements is consistent with the change in TS 3.3.4's Applicability. The revised Action statement reflects that the Containment purge valve isolation signal is not required by the FHA analyses but is still required to meet the operability requirements for the containment purge valves in MODES 1, 2 , 3, and 4. The NRC staff finds that the proposed change is acceptable.

The effect of the proposed changes to these TSs is to allow plant operation without the capability for an automatic purge valve closure during core alterations and movement of irradiated fuel. In performing the re-analyses of the FHAs and spent fuel cask drop accidents, DNC did not credit any filtration, holdup, or dilution prior to release to the environment during a DBA FHA. The events were conservatively modeled as if there was no enclosing structure (CNMT or spent fuel building). Since the operability of the design feature is no longer assumed as an initial condition in a DBA analysis, the requested changes are acceptable with regard to DBA radiological consequences.

3.4.3 TS 3.7.6.1 Plant Systems, Control Room Emergency Ventilation System

3.4.3.1 The licensee proposes to make changes to the Applicability statement so that the LCO is no longer applicable in Modes 5 and 6 and during movement of a shielded cask over the spent fuel pool cask lay down area. Additionally, the licensee proposes to modify the statement “During fuel movement within containment or the spent fuel pool.” to add “irradiated” in front of fuel. This proposal was modified to retain applicability in Modes 5 and 6 in the May 7, 2004, supplemental letter.

The current TS is applicable in Modes 5 and 6 primarily for CORE ALTERATIONS, fuel movement, and the movement of a shielded cask over the spent fuel pool cask lay down area. In the revised FHA analyses, the limiting event is an FHA which cannot occur in Mode 5 as irradiated fuel cannot be moved in that mode. The TS continues to be applicable during the movement of irradiated fuel, therefore, the deletion of Modes 5 and 6 from the Applicability statement is acceptable. The NRC staff expressed concern that another unit of this multiple unit site could be in operation during Modes 5 and 6 and that the Control Room Emergency Ventilation System (CREVS) should be operable to provide protection from events at the other operating unit. The licensee modified its submittal to include Modes 5 and 6 in the applicability.

The licensee states that the movement of a spent fuel cask over the spent fuel pool cask lay down area can be deleted since the revised FHA analyses do not postulate the shielded cask drop as a limiting case accident. The FHA analyses only credit the operation of CREVS during an FHA with irradiated fuel in the containment or the spent fuel pool. Since the FHA analyses take no credit for CREVS in the event of dropping a spent fuel cask over the spent fuel pool cask lay down area will not result in a DBA, the NRC staff finds change in applicability is acceptable.

The addition of “irradiated” in front of fuel in the Applicability statement “During fuel movement within containment or the spent fuel pool.” is consistent with the revised FHA analysis that does not require CREVS operation in the event of an FHA involving new fuel, therefore the proposed change is acceptable.

3.4.3.2 The licensee proposes the following changes to the ACTION statements:

3.4.3.2.1 In Action statements b., c., d. and e., the word “irradiated” is being placed in front of “fuel assemblies” and the phrase “...and the movement of shielded casks over the spent fuel pool cask lay down area” is being deleted.

The NRC staff finds these changes acceptable as they are consistent with the change in Applicability discussed above in Section 3.3.1.

3.4.3.2.2 The phrase “MODES 5 and 6, and all other times” prior to Action statements d. and e. will be replaced with “During irradiated fuel movement within containment or the spent fuel pool.” In the marked pages of TS changes provided with the May 7, 2004, letter, the licensee revised the replacement to read “Modes 5 and 6 and During irradiated fuel movement within containment or the spent fuel pool.”

The NRC staff finds this change to be acceptable. Including Modes 5 and 6 provides protection for control room operators from events at other operating units on the site even though the FHA analysis shows that the CREVS is not necessary for a refueling accident.

3.4.3.2.3 In action statements b, c, d, and e, the phrase “and the movement of shielded casks over the spent fuel pool cask lay down area.” is being deleted.

The licensee states that the revised FHA analyses do not postulate a shielded cask drop as a limiting case accident and only requires that the CREVS be operational when irradiated fuel is being moved in the containment or spent fuel pool. The NRC staff concurs that this phrase may be deleted. Although operability is not required to be controlled by the TS, the licensee should consider that the operability of this system may be desirable in satisfying GDC (or corresponding proposed GDC) requirements or minimizing potential releases.

The effect of this proposed change is to allow plant operation without the capability for control room isolation and recirculation filtration during movement of new fuel or during the movement of a shielded cask over the spent fuel pool cask lay down area. No credit was taken for the operation of the CREVS in the spent fuel cask drop accident analysis. The consequences of an FHA involving new fuel would be negligible and well below the consequences of an FHA involving irradiated fuel. Credit was taken in the analyses of FHAs involving irradiated fuel and the proposed change language retains the operability requirement for this condition. As such, the proposed changes to this TS are acceptable with regard to DBA radiological consequences.

3.4.4 TS 3.9.4 Refueling Operations, Containment Penetrations

3.4.4.1 The licensee, as part of the proposed changes discussed below, will add a note defining the required administrative controls necessary for an equipment door, as well as any other penetration, to be opened during movement of irradiated fuel assemblies which will read:

> * Administrative controls shall ensure that appropriate personnel are aware that the equipment door, personnel air lock door and/or other containment penetrations are open, and that a specific individual(s) is designated and available to close the equipment door, personnel airlock door and/or other containment penetrations within 30 minutes if a fuel handling accident occurs. Any obstructions (e.g. cables and hoses) that could prevent closure of the equipment door, a personnel air lock door and/or other containment penetration must be capable of being quickly removed.

The NRC staff concurs that having the containment penetrations open during refueling reduces defense-in-depth and that closing the penetrations provides an additional measure of protection to the public. The administrative controls (proposed by the licensee as a footnote to TS 3.9.4) to close the containment in 30 minutes with designated personnel after an FHA compensate for the reduction in defense-in-depth. The NRC staff has determined that these administrative controls provide an important element of defense-in-depth and assure that the licensee will manage the consequences of an FHA in a manner that will afford adequate protection to the public.

3.4.4.2 The licensee proposes to change LCO a. from “The equipment door closed and held in place by a minimum of four bolts,” to “The equipment door closed or capable of being closed under administrative control,*”. In the May 7, 2004, supplemental letter, the licensee revised the phrase, “The equipment door closed or capable of being closed under administrative control,*” to state that “The equipment door shall be either: 1. closed and held in place by a minimum of four bolts, or 2. open under administrative control* and capable of being closed and held in place by a minimum of four bolts.”

The licensee states that the “revised FHA Analyses assumes that all of the radioactive material which could be released to the containment atmosphere exits the containment within two (2) hours of accident initiation with no credit taken for the containment boundary closure.” As a consequence, the containment isolation is no longer on the primary success path of a DBA. Thus, the requirement for an LCO based on Criterion 3 of 10 CFR 50.36 does not apply.

The licensee proceeds to state that “Consistent with the philosophy of minimizing dose released to the environment, administrative controls will be established to ensure that the equipment access hatch, and other containment penetrations which provide direct access to the outside atmosphere, can be closed within 30 minutes of accident initiation as a defense-in-depth measure to minimize the consequences of a[n] FHA.”

The licensee states that the “containment atmosphere is monitored during normal and transient operations of the reactor plant by the containment structure particulate and gas monitor located in the upper level of the Auxiliary Building or by grab sampling.” The licensee also states that since this proposed change will allow containment penetrations to be open under administrative control for extended periods of time during refueling outages, routine grab samples of the containment atmosphere, the equipment access hatch and personnel access hatch will be required.” The NRC staff confirms that the use of existing monitors along with the use of grab samples taken at the appropriate locations would provide sufficient monitoring to comply with the provisions of proposed GDC 17 (similar to GDC 64).

The NRC staff also considered the implications of the proposed change on GDC 61 (or corresponding proposed GDC) which requires appropriate containment, confinement, and filtering of radioactive contaminants in areas where fuel is stored. The staff considers the licensee’s commitment to administrative controls which close the equipment hatch, terminate the purge, and isolate the containment as satisfying the requirements of proposed GDCs 69 and 70 (the combination of which is similar to GDC 61) and minimize any potential release to the public.

The NRC staff concurs that having the containment penetrations open during refueling reduces defense-in-depth and that closing the penetrations provides an additional measure of protection to the public. The administrative controls (proposed by the licensee as a footnote to TS 3.9.4) to close the containment in 30 minutes with designated personnel after an FHA compensate for the reduction in defense-in-depth. The NRC staff has determined that these administrative controls provide an important element of defense-in-depth and assure that the licensee will manage the consequences of an FHA in a manner that will afford adequate protection to the public.

3.4.4.3 The licensee proposes to change LCO b.2 with regards to the personnel air lock being open:

Existing b.2.: capable of being closed by an OPERABLE personnel air lock door, under administrative control, with containment purge in operation, and

Proposed b.2.: capable of being closed by an OPERABLE personnel air lock door, under administrative control*, and

The NRC staff finds that adding the "*" and the appropriate footnote to define the administrative controls which specify closure requirements is acceptable and enhances the clarity of the TSs.

The NRC staff finds deleting the phrase "with containment purge in operation" is acceptable with the following clarification. The licensee states that since the revised FHA analysis does not make any assumptions as to containment purge operation during an FHA, the phrase can be deleted. It should be noted that the containment purge is not automatically isolated during an FHA by changes proposed in this submittal. Thus, the containment purge may be operating or it may have been manually isolated. In either case, the personnel air lock door must be capable of being closed. If the air flow through the personnel air lock is sufficiently large to impede the closure of an air lock door, the licensee must address this condition prior to making the change.

3.4.4.4 The licensee proposes to change LCO c.2 concerning penetrations:

Existing c.2.: Be capable of being closed by an OPERABLE Containment Purge Valve Isolation System.

Proposed c.2.: Be capable of being closed under administrative controls*

The revised FHA analyses do not take credit for the automatic closure of the containment purge valves or for containment boundary integrity. Closing penetrations with direct access from the containment atmosphere to the outside atmosphere is a mitigating action that reduces the release due to the FHA. The NRC staff finds that controlling the penetrations using the administrative controls which are defined in the footnote "*" provides additional mitigation of an FHA and defense-in-depth. As such, the change is acceptable.

3.4.4.5 The licensee proposes to delete CORE ALTERATIONS from the LCO applicability.

The NRC staff's position is that the FHA is the only event during CORE ALTERATIONS that is postulated to result in fuel damage and radiological release. The proposed change to the applicability statement leaves the LCO and Required Actions applicable during activities which could result in an FHA with fuel damage and radiological release. Therefore, the NRC staff finds that the deletion of CORE ALTERATIONS is acceptable.

3.4.4.6 The licensee proposes to delete SR 4.9.4.2 which requires verification that each containment purge valve actuates on an actual or simulated actuation signal at least once per 18 months.

The proposed change is consistent with the revised FHA analyses that does not take credit for the automatic closure containment purge valves in the event of an FHA. Thus, the NRC staff finds that the deletion of SR 4.9.4.2 is acceptable.

The effect of these proposed changes is to allow movement of irradiated fuel inside CNMT with the equipment door open and without the capability of automatic isolation of CNMT purges. In performing the re-analyses of the FHAs and spent fuel cask drop accidents, DNC did not credit any filtration, holdup, or dilution prior to release to the environment. Since the operability of the design features is no longer assumed as an initial condition in a DBA analysis, the requested changes are acceptable with regard to DBA radiological consequences.

3.4.5 TS 3.9.8.1 Refueling Operations, Shutdown Cooling and Coolant Circulation - High Water Level

Item 3.a.c.3) of the Note associated with the LCO allows the shutdown cooling pumps to be removed from service for the time required to perform local leak rate testing of containment penetration number 10 or to permit maintenance on valves on the common shutdown cooling suction line provided, among other conditions, the containment penetrations are in the status outlined below. Action statement c.3. states that in the event of no shutdown cooling train OPERABLE or in operation, the containment penetrations should be placed in the same status as Item 3.a.c.3) within 4 hours. The licensee proposes to modify these requirements which currently read:

> Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:
>
> a) Closed by a manual or automatic isolation valve, blind flange, or equivalent, or
>
> b) Be capable of being closed by the Containment Purge Valve Isolation System.

The proposed change replaces the above in the LCO Note and the Action statement with the following:

> Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be closed by a manual or automatic isolation valve, blind flange, or equivalent.

The revised FHA analyses do not take credit for the automatic closure of the containment purge valves or for containment boundary integrity. Closing penetrations with direct access from the containment atmosphere to the outside atmosphere is a mitigating action that reduces the release due to the FHA. The proposed changes are more conservative than the existing requirements since the option to have penetrations open but capable of being closed by the containment purge isolation system is removed. Since the revised LCO and Action statement will place the valves in a conservative status, the NRC staff determined that the proposed changes are acceptable.

The effect of this change is to omit the CNMT purge valve isolation system as a means of isolating CNMT penetrations. In performing the re-analyses of the FHA accidents, DNC did not credit any filtration, holdup, or dilution prior to release to the environment during a DBA FHA. Since the operability of this design feature is no longer assumed as an initial condition in a DBA analysis, the requested changes are acceptable with regard to DBA radiological consequences.

3.4.6 TS 3.9.8.2, Refueling Operations, Shutdown Cooling and Coolant Circulation - Low Water Level

The licensee proposes to modify Action statement c.3. which currently reads:

> Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:
>
> a) Closed by a manual or automatic isolation valve, blind flange, or equivalent, or
>
> b) Be capable of being closed by an OPERABLE Containment Purge Valve Isolation System.

The proposed Action statement reads:

> Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be closed by a manual or automatic isolation valve, blind flange, or equivalent.

The proposed changes are more conservative than the existing requirements since the option to have penetrations open but capable of being closed by the containment purge isolation system is removed. Since the revised Action statement will place the valves in a conservative status, the NRC staff determined that the proposed changes are acceptable.

The effect of this change is to omit the CNMT purge valve isolation system as a means of isolating CNMT penetrations. In performing the re-analyses of the FHA accidents, DNC did not credit any filtration, holdup, or dilution prior to release to the environment during a DBA FHA. Since the operability of this design feature is no longer assumed as an initial condition in a DBA analysis, the requested changes are acceptable with regard to DBA radiological consequences.

3.4.7 TS 3.9.15 Refueling Operations, Storage Pool Area Ventilation System.

The licensee has shown on the basis of its FHA design basis analysis that the fuel building exhaust filter system is not required to satisfy the dose guidelines of 10 CFR 50.67. Thus, the system is not on the primary success path for a DBA. As such, Criterion 3 of 10 CFR 50.36 does not require an LCO to be placed in the TSs. The fuel handling DBA does have an assumption as to the time at which the accident occurs. The licensee is restricted from moving the fuel before this time based upon the requirements of its decay heat time TS 3/4.9.3 which limits the movement of irradiated fuel until the reactor has been subcritical for 150 hours. Since movement of fuel prior to this time is restricted by the TSs, the inclusion of an LCO to satisfy the requirements of Criterion 2 of 10 CFR 50.36 which requires an LCO for process variables upon which a design basis analysis depends is not required. The NRC staff concurs that the licensee may remove the section from its proposed TSs.

The NRC staff's finding of acceptability for this item is only associated with the removal of the item from the TSs. Prior to removing any equipment or changing any procedure affecting the operation of engineering safeguards equipment, the licensee must use the appropriate modification process (10 CFR 50.59 or 10 CFR 50.90) to assure that the facility complies with all other commitments including GDC (or corresponding proposed GDC) as stated in 10 CFR Part 50 Appendix A, or their equivalents, that the appropriate changes are made to the facility UFSAR and that defense-in-depth and safety margins are adequate. The licensee has stated that "procedural guidance will be available for closing fuel building area atmosphere boundary penetrations if a[n] FHA occurs inside the fuel building." The use of this procedural guidance will be implemented "as a defense in depth measure to minimize actual releases to the outside atmosphere much lower than assumed in the AST FHA analyses dose calculations." The NRC staff concurs that the development and implementation of procedural guidance will increase defense-in-depth and facilitate managing releases during an FHA. As such, it will provide increased protection to public health and safety.

The effect of this proposed change is to allow movement of irradiated fuel without the need to satisfy current requirements for spent fuel area integrity, storage pool area ventilation system, and operability of an enclosure building filtration train. In performing the re-analyses of the FHAs and spent fuel cask drop accidents, DNC did not credit any filtration, holdup, or dilution prior to release to the environment during a DBA FHA. Since the operability of the design features is no longer assumed as an initial condition in a DBA analysis, the requested changes are acceptable with regard to DBA radiological consequences.

4.0 SUMMARY

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by DNC to assess the radiological impacts of the proposed license amendment at MP2. The staff finds that DNC 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 DNC to the applicable criteria identified in Section 2.0. The staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and control room doses will continue to comply with these criteria. Therefore, the proposed license amendment is acceptable with regard to the radiological consequences of postulated DBAs.

This licensing action is considered a selective implementation of the AST. With this approval, the selected characteristics of the AST and TEDE criteria become the design basis for the DBA FHA within the CNMT and outside the CNMT. This approval is limited to this specific implementation. Subsequent modifications, based on the selected characteristics incorporated into the design basis by this action, may be possible under the provisions of 10 CFR 50.59. However, use of other characteristics of an AST, and changes to previously approved AST characteristics, requires prior NRC staff approval pursuant to 10 CFR 50.67. The selected characteristics of the AST and the TEDE criteria may not be extended to other aspects of the plant design or operation without prior NRC review, pursuant to 10 CFR 50.67. All future FHA radiological analyses performed to demonstrate compliance with regulatory requirements shall address the selected characteristics of the AST and the TEDE criteria as described in the MP2 design basis.

The proposed changes to the TSs identified in Section 3.4 were reviewed by the NRC staff and found to be in compliance with NRC's regulations. Thus, the licensee may implement these changes to their TSs. The licensee has the responsibility to evaluate any modifications to plant configuration they make as a result of the TS revisions to its plant equipment, operating procedures, and surveillance programs. This evaluation must assure compliance with design criteria such as the GDC (or corresponding proposed GDC) in Appendix A or 10 CFR Part 50 or their equivalents, the UFSAR, and other plant commitments and demonstrate that safety margins and that defense-in-depth are maintained. The licensee’s submittal demonstrates that the FHA analyses meets the requirements of 10 CFR 50.67. This compliance, along with 10 CFR 50.36, establishes the regulatory basis upon which the TS changes can be made.

The NRC staff acknowledges receipt of the conforming changes that were made to the TS Bases as provided in the licensee’s letter dated September 26, 2002.

5.0 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 had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 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 (68 FR 40711). 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.

7.0 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.

Principal Contributors: S. La Vie
E. Forrest
U. Shoop
V. Nerses

Date: September 20, 2004

Table 1

Millstone Unit 2 Accident Analysis Parameters

All CASES

Reactor power (2700 x 1.02), MWt	2754
Core peaking factor	1.83
Dose conversion factors	FGR11/FGR12
Breathing rate, offsite, m^3/s	
0-8 hours	3.5E-4
8-24 hours	1.8E-4
>24 hours	2.3E-4
Breathing rate, control room, m^3/s	3.5E-4
Control room normal intake flow, cfm	800
Control room isolation delay, sec	10
Control room shift to filtered recirculation, minutes	10
Control room unfiltered infiltration, cfm	130
Control room filtered recirculation, cfm	2250
Control room charcoal filter efficiency, %	90
Control room volume, ft^3	35,650
Control room occupancy factor	
0-24 hrs	1.0
1-4 days	0.6
4-30 days	0.4

Fuel Handling Accident*

Fuel assemblies damaged	1
Decay period, hrs	100
Fraction of core in gap	
Iodine-131	0.12
Kr-85	0.30
Other iodines	0.10
Other noble gases	0.10
Pool decontamination factor	200

Release period	100% over 2 hours
Hold-up & release mitigation	No credit taken
Release via:	Enclosure building edge
Control room isolation, sec	20
Atmospheric dispersion, sec/m^3	
EAB	3.66E-4
LPZ	4.80E-5
Control Room	5.46E-3
Release chemical form, percent	
Elemental	57
Organic	43

** Applicable to FHA inside CNMT and inside spent fuel pool area*

Spent Fuel Cask Drop Accident*

Fuel assemblies damaged	
One year decay	184
Five-year decay	1376
Fraction of core in gap	
I-129	0.10
Kr-85	0.30
Pool decontamination factor	200
Release period	100% over 2 hours
Hold-up & release mitigation	No credit taken
Release via:	Enclosure building edge
Control room isolation, sec	No credit taken
Atmospheric dispersion, sec/m^3	
EAB	3.66E-4
LPZ	4.80E-5
Control Room	5.46E-3
Release chemical form, percent	
Elemental	57
Organic	43