UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 15, 2009

Mr. Stewart B. Minahan
Vice President-Nuclear and CNO
Nebraska Public Power District
72676 648A Avenue
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT RE: APPLICATION OF THE ALTERNATIVE SOURCE TERM FOR LOSS-OF-COOLANT ACCIDENT DOSE CONSEQUENCES (TAC NO. MD9921)

Dear Mr. Minahan:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 234 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated October 13, 2008, as supplemented by letters dated April 8, May 29, June 12, and September 1, 2009.

The amendment would revise the licensing basis by approving adoption of the Alternative Source Term (AST), in accordance with Section 50.67, "Accident source term," of Title 10 of the *Code of Federal Regulations* (10 CFR), for use in calculating the loss-of-coolant accident (LOCA) dose consequences. The amendment would revise the TSs to (1) change the TS definition for DOSE EQUIVALENT I-131 to adopt Federal Guidance Report 11 dose conversion factors; (2) require operability of the Standby Liquid Control system in Mode 3, to reflect its credit in the LOCA analysis; (3) establish a Main Steam (MS) Pathway leakage limit that effectively increases the previous MS isolation valve leakage limit; and (4) change TS Section 5.5.12 to reflect a requested permanent exemption from the requirements of 10 CFR Part 50, Appendix J, Option B, Section III.A, to allow exclusion of MS Pathway leakage from the overall integrated leakage rate measured during the performance of a Type A test, and from the requirements of Appendix J, Option B, Section III.B, to allow exclusion of the MS Pathway leakage from the combined leakage rate of the penetrations and valves subject to Type B and C tests. The requested exemption will be issued via separate correspondence.

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

Sincerely,

C F Lyon

Carl F. Lyon, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures:
1. Amendment No. 234 to DPR-46
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 234
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District (the licensee), dated October 13, 2008, as supplemented by letters dated April 8, May 29, June 12, and September 1, 2009, 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 license 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-46 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A as revised through Amendment No. 234, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented within 45 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility
Operating License No. DPR-46
and Technical Specifications

Date of Issuance: September 15, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 234

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of the Facility Operating License No. DPR-46 and Appendix A Technical Specifications with the enclosed revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Operating License

REMOVE	INSERT
Page 3 of 5	Page 3 of 5

Technical Specifications

REMOVE	INSERT
1.1-2	1.1-2
1.1-3	1.1-3
3.1-20	3.1-20
3.6-14	3.6-14
3.6-15	3.6-15
5.0-16	5.0-16
5.0-17	5.0-17

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2419 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 234, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Cooper Nuclear Station Safeguards Plan," submitted by letter dated May 17, 2006.

(4) Fire Protection

The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Cooper Nuclear Station (CNS) Updated Safety Analysis Report and as approved in the Safety Evaluations dated November 29, 1977; May 23, 1979; November 21, 1980; April 29, 1983; April 16, 1984; June 1, 1984; January 3, 1985; August 21, 1985; April 10, 1986; September 9, 1986; November 7, 1988; February 3, 1989; August 15, 1995; and July 31, 1998, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

1.1 Definitions

CHANNEL CHECK (continued)	status derived from independent instrument channels measuring the same parameter.
CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and b. Control rod movement, provided there are no fuel assemblies in the associated core cell. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that would produce the same dose as the quantity and isotopic mixture of I-131, I-132,

(continued)

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)

I-133, I-134, and I-135 actually present. The DOSE EQUIVALENT I-131 concentration is calculated as follows: DOSE EQUIVALENT I-131 = (I-131) + 0.0060 (I-132) + 0.17 (I-133) + 0.0010 (I-134) + 0.029 (I-135). The dose conversion factors used for this calculation are those listed in Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989.

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or

2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE;

d. Pressure Boundary LEAKAGE

LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

LOGIC SYSTEM FUNCTIONAL TEST

A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components required for OPERABILITY of a logic circuit,

(continued)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SLC subsystem inoperable.	A.1 Restore SLC subsystem to OPERABLE status.	7 days
B. Two SLC subsystems inoperable.	B.1 Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify the isolation time of each MSIV is $\geq$ 3 seconds and $\leq$ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	18 months
SR 3.6.1.3.8	Verify a representative sample of reactor instrumentation line EFCVs actuate to the isolation position on an actual or simulated instrument line break.	18 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	18 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	Verify leakage rate through each Main Steam line is $\leq$ 106 scfh when tested at $\geq$ 29 psig.	In accordance with the Primary Containment Leakage Rate Testing Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.11	Verify each inboard 24 inch primary containment purge and vent valve is blocked to restrict the maximum valve opening angle to 60°.	18 months
SR 3.6.1.3.12	Verify leakage rate through the Main Steam Pathway is $\leq$ 212 scfh when tested at $\geq$ 29 psig.	In accordance with the Primary Containment Leakage Rate Testing Program

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to support system(s) for the supported systems b.1 and b.2 above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

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

    1. Exemption from Appendix J to 10CFR Part 50 to allow reverse direction local leak rate testing of four containment isolation valves at Cooper Nuclear Station (TAC NO. M89769) (July 22, 1994).

    2. Exemption from Appendix J to 10CFR Part 50 to allow MSIV testing at 29 psig and expansion bellows testing at 5 psig between the plies (Sept. 16, 1977).

    3. Exception to NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Section 9.2.3: The first Type A test performed after the December 7, 1998 Type A test shall be performed no later than December 7, 2013.

    4. Exemption from Section III.A of 10CFR Part 50, Appendix J, Option B, to allow the leakage contribution from Main Steam Pathway (Main Steam lines and Main Steam inboard drain line) leakage to be excluded from the overall integrated leakage rate from Type A tests (September 14, 2009).

(continued)

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

5. Exemption from Section III.B of 10CFR Part 50, Appendix J, Option B, to allow the contribution from Main Steam Pathway (Main Steam lines and Main Steam inboard drain line) leakage to be excluded from the sum of the leakage rates from Type B and Type C tests (September 14, 2009).

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 58.0 psig. The containment design pressure is 56.0 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.635% of containment air weight per day.

d. Leakage Rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are, <0.60 L_a for the Type B and C tests and $\leq$ 0.75 L_a for Type A tests.

2. Air lock testing acceptance criteria are:

   a. Overall air lock leakage rate is $\leq$ 12 scfh when tested at $\geq P_a$.

   b. Overall air lock leakage rate is $\leq$ 0.23 scfh when tested at $\geq$ 3.0 psig.

e. The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

f. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

5.5.13 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Filter (CREF) System, CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without

(continued)

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 234 TO

FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By application dated October 13, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML082910760), as supplemented by letters dated April 8, May 29, June 12, and September 1, 2009 (ADAMS Accession Nos. ML091040565, ML091530354, ML091671811, and ML092510169, respectively), Nebraska Public Power District (NPPD, the licensee), requested changes to the Technical Specifications (TSs) for Cooper Nuclear Station (CNS).

The proposed changes would revise the licensing basis by approving adoption of the Alternative Source Term (AST), in accordance with Section 50.67, "Accident source term," of Title 10 of the *Code of Federal Regulations* (10 CFR), for use in calculating the loss-of-coolant accident (LOCA) dose consequences. The amendment would revise the TSs to (1) change the TS definition for DOSE EQUIVALENT I-131 to adopt Federal Guidance Report (FGR) 11 dose conversion factors; (2) require operability of the Standby Liquid Control (SLC) system in Mode 3, to reflect its credit in the LOCA analysis, (3) establish a Main Steam (MS) Pathway leakage limit that effectively increases the previous MS isolation valve leakage limit; and (4) change TS Section 5.5.12 to reflect a permanent exemption from the requirements of 10 CFR Part 50, Appendix J, Option B, Section III.A, to allow exclusion of MS Pathway leakage from the overall integrated leakage rate measured during the performance of a Type A test, and from the requirements of Appendix J, Option B, Section III.B, to allow exclusion of the MS Pathway leakage from the combined leakage rate of the penetrations and valves subject to Type B and C tests.

With its amendment request, the licensee requested an exemption from (1) the requirements of 10 CFR Part 50, Appendix J, Option B, Section III.A to allow exclusion of the main steam line leakage pathway (including the leakage from the main steam inboard drain line) from the overall integrated leakage rate measured when performing a Type A test; and (2) the requirements of

10 CFR 50, Appendix J, Option B, Section III.B, to allow exclusion of the main steam line leakage pathway (including the leakage from the main steam inboard drain line) from the combined leakage rate of the penetrations and valves subject to Type B and C tests. The requested exemption will be issued via separate correspondence from the U.S. Nuclear Regulatory Commission (NRC).

The supplemental letters dated April 8, May 29, June 12, and September 1, 2009, 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 January 23, 2009 (74 FR 4251).

Specifically, NPPD proposes to revise the licensing basis of the LOCA described in Section XIV-6.3 of the CNS updated safety analysis report (USAR). The proposed licensing basis change is to use the AST methodology for dose consequences analysis in accordance with 10 CFR 50.67, NRC Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (ADAMS Accession No. ML003716792), and NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," dated February 1995. The licensee will revise the USAR as an implementing action following issuance of the license amendment, with revised USAR pages submitted pursuant to 10 CFR 50.71(e).

Consistent with the proposed change to the licensing basis, the licensee proposes the following TS changes:

1. TS 1.1, "Definitions," is revised to change the definition of DOSE EQUIVALENT I-131 to reflect the dose conversion factors contained in Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

2. TS 3.1.7, "Standby Liquid Control (SLC) System," is revised by adding MODE 3 to the APPLICABILITY, and by adding Required Action C.2 for CNS to be in MODE 4, with a Completion Time of 36 hours.

3. In TS 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," Surveillance Requirement (SR) 3.6.1.3.10 currently states:

   > Verify combined main steam leakage rate is ≤ 46 scfh when tested at ≥ 29 psig.

   This SR is revised to state:

   > Verify leakage rate through each Main Steam line is ≤ 106 scfh when tested at ≥ 29 psig.

4. Also in TS 3.6.1.3, new SR 3.6.1.3.12 is added to state:

   Verify leakage through the Main Steam Pathway is ≤ 212 scfh when tested at ≥ 29 psig.

   The Frequency is specified as "In accordance with the Primary Containment Leakage Rate Testing Program."

5. In TS 5.5.12, "Primary Containment Leakage Rate Testing Program," paragraph a, exception numbers 4 and 5 discuss exemptions from Section III.A and III.B, respectively, from 10 CFR Part 50, Appendix J, Option B. The licensee proposes to revise these exceptions by replacing "MSIV" with "Main Steam Pathway (Main Steam lines and the Main Steam inboard drain line)." Also, the date of the approved exemption in parentheses is to be revised from "October 30, 2006" to the date that the proposed exemption is approved.

The licensee also proposes conforming revisions to the associated TS Bases as part of the implementation of the amendment, to be made in accordance with TS 5.5.10, "Technical Specifications Bases Control Program."

2.0 BACKGROUND

In December 1999, the NRC issued a new regulation, 10 CFR 50.67, "Accident source term," which provided a mechanism for licensees of power reactors to voluntarily replace the traditional accident source term used in their design-basis accident (DBA) analyses with an AST. The "source term" referred to throughout this evaluation refers to the fission product release from the reactor core into containment during a DBA. The source term is characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release from the core following DBAs. Examples of accidents for which radiological consequence analyses are performed for the purposes of calculating the source term include a LOCA, main steam line break, fuel handling accident, and control rod drop accidents. Regulatory guidance for the implementation of the AST is provided in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." The regulations in 10 CFR 50.67 require a licensee seeking to use an AST to apply for a license amendment and require that the application contain an evaluation of the consequences of DBAs.

A selective-scope AST implementation refers to the licensee's request to recalculate the dose consequences of select DBAs as opposed to the full-scope implementation, which requires the recalculation of dose consequences for all of the DBAs mentioned above. NPPD proposes to selectively apply the requirements and guidance to use an AST in evaluating the offsite and control room radiological consequences of a LOCA. This reanalysis involves several changes in selected-analysis assumptions including different atmospheric dispersion values for the control room outside air intake. As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance." It also replaces the whole body (and its equivalent to any part of the body) dose criteria of 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19, "Control room."[1]

3.0 EVALUATION

3.1 Radiological Consequences Analysis

The current CNS licensing basis uses a source term that is based on Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," U.S. Atomic Energy Commission, March 23, 1962 (ADAMS Legacy Library Accession No. 8202010067), to calculate the radiological consequences of the postulated LOCA. NPPD's license amendment request (LAR) contains the reanalysis and a description of the licensing basis alternative method. Use of the AST methodology allows licensees to increase the assumed design-basis unfiltered inleakage into the control room envelope to a value larger than that empirically observed, thereby providing additional margin in plant design.

3.1.1 Regulatory Evaluation

The NRC staff evaluated the licensee's analysis of the radiological consequences of the postulated LOCA against the dose acceptance criteria specified in 10 CFR 50.67. The applicable criteria are 5 roentgen equivalent man (rem) TEDE in the control room for the duration of the event, 25 rem TEDE at the exclusion area boundary (EAB) for the worst 2 hours, and 25 rem TEDE at the outer boundary of the low population zone (LPZ) for the duration of the event. The dose acceptance criterion in vital areas is accepted to be 5 rem TEDE for the duration of the accident to show compliance with the regulatory requirements of NUREG-0737 and Section IV.E.8 of Appendix E to 10 CFR Part 50.

The regulatory requirements upon which the NRC staff based its acceptance are those in GDC 19, and the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 and Table 6 of RG 1.183 and NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 15.0.1. The licensee has not proposed any deviation or departure from the guidance provided in RG 1.183. The NRC staff's evaluation is based upon the following regulatory codes, guides, and standards, in addition to relevant information in the CNS USAR and TSs, as well as consideration for any applicable alternative documentation the licensee may have provided:

- 10 CFR 50.67, "Accident source term."
- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," GDC 19, "Control room."
- 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance."

[1] The 1967 Proposed GDC as described in the CNS updated safety analysis report (USAR), Appendix F, are the licensing basis for CNS; however, the NRC staff concluded in its 1973 Safety Evaluation Report for CNS that the intent of the 1971 Final Rule for 10 CFR Part 50, Appendix A, had also been met.

- NUREG-0737, “Clarification of TMI Action Plan Requirements.”
- NUREG-0800, SRP Section 2.3.4, “Short-Term Atmospheric Dispersion Estimates for Accident Releases.”
- NUREG-0800, SRP Section 6.4, “Control Room Habitability System.”
- NUREG-0800, SRP Section 6.5.2, “Containment Spray as a Fission Product Cleanup System.”
- NUREG-0800, SRP Section 15.0.1, “Radiological Consequence Analyses Using Alternative Source Terms.”
- NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants.”
- NUREG/CR-5732, “Iodine Chemical Forms in LWR Severe Accidents.”
- NUREG/CR-5950, “Iodine Evolution and pH Control.”
- RG 1.23, “Meteorological Monitoring Programs for Nuclear Power Plants.”
- RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”
- RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants.”
- RG 1.197, “Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors.”

3.1.2 Technical Evaluation

The NRC staff reviewed the technical analyses performed by the licensee in support of its proposed license amendment, as related to the radiological consequences of the design-basis LOCA analysis. Information regarding this analysis was provided in Enclosure 1 of the licensee's application dated October 13, 2008. The staff reviewed the assumptions, inputs, and methods used by the licensee to assess the impacts of the proposed license amendment. The staff also performed independent calculations to confirm the conservatism of the licensee’s analysis. The findings of this safety evaluation (SE) are based on the descriptions and results of the licensee’s analysis and other supporting information submitted by the licensee.

3.1.2.1 Atmospheric Dispersion Estimates

3.1.2.1.1 Meteorological Data

NPPD used hourly onsite meteorological data collected from 1994 through 1998 as input to the ARCON96 computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes") to calculate control room atmospheric dispersion factors (χ/Q values) other than for fumigation conditions assumed to occur from the stack. The data were submitted in support of prior LARs and, as formatted for the current LAR, were provided by letter dated January 24, 2003 (ADAMS Accession No. ML030290389). Meteorological measurements used in these calculations were made at the 10, 60, and 100 meter levels.

SEs associated with CNS Amendment No. 183 (ADAMS Accession No. ML003700347), dated April 7, 2000, Amendment No. 187 (ADAMS Accession No. ML012960618), dated October 23, 2001, Amendment No. 196 (ADAMS Accession No. ML030560804), dated February 21, 2003, and Amendment No. 222 (ADAMS Accession No. ML062260239), dated September 5, 2006; discuss the 1994 through 1998 data and their previous applications in generating control room χ/Q values. Of particular note with respect to the current LAR, for the 5-year period, joint wind speed, wind direction, and atmospheric stability data recovery was less than the recommended minimum of 90 percent cited in RG 1.23. NRC staff review of the data indicated that this was primarily due to lower data recovery in 1995 and 1996 with respect to the temperature difference measurements as a function of height (Δ-T), which were used to determine atmospheric stability and, to a lesser extent, some of the wind direction measurements. NRC staff noted a few variations in the atmospheric stability measurements, which the licensee attributed to factors such as climatologic variability, wind shifts, and minor temperature fluctuations. As part of Amendment No. 196, the licensee provided a description of a series of measures initiated in 1998 to improve the meteorological measurements program. The NRC staff determined that the 1994 through 1998 data are adequate for use in the dose assessments associated with the current LAR, but recommends that data from the improved program be considered for use in any future calculations.

3.1.2.1.2 Control Room Atmospheric Dispersion Factors

The licensee previously calculated ground level and elevated release χ/Q values for control room dose assessments using the 1994 through 1998 meteorological data as discussed in the SEs associated with CNS Amendment Nos. 183, 187, 196, and 222. As noted in the SE associated with CNS Amendment No. 187, initially the licensee made χ/Q estimates for each of the 5 years individually using the ARCON96 computer code and selected the highest values irrespective of year. When a data processing error was subsequently discovered, the licensee recalculated χ/Q values using the entire 5-year interval, compared the results with the previously calculated year-by-year estimates, and used the higher values in its dose assessment. In the current LAR, the licensee has used the control room χ/Q values based upon the entire 1994 through 1998 meteorological data period, with the corrected data processing. The licensee also revised the temporal distribution of the elevated release control room χ/Q values to conservatively model fumigation of the elevated release to occur during the 2-hour interval of maximum release to the environment, beginning at 1.3 hours into the event. These control room χ/Q values are listed in Tables 3.1.1 and 3.1.2 of this SE.

As noted in Section 3.1.2.1.1 above, NRC staff continued to have some concern about the quality of the data used in the current LAR, particularly with regard to the lower data recovery in 1995 and 1996. Therefore, staff generated a comparison set of $\chi/_Q$ values based upon the 1994, 1997, and 1998 3-year period when recovery each year was greater than 90 percent as recommended in RG 1.23. The staff found that the $\chi/_Q$ values based upon the 3-year period were not significantly different than those generated by the licensee using the 5-year data period in this specific case and, therefore, are acceptable. However, the NRC staff recommends that data from the improved program be considered for use in any future calculations.

3.1.2.1.3 Offsite Atmospheric Dispersion Factors

NPPD used previously calculated ground level and elevated release $\chi/_Q$ values as listed in Tables 3.1.1 and 3.1.2 of this SE for inputs to the exclusion area boundary (EAB) and low population zone (LPZ) dose assessment. These values were previously approved by the NRC, as discussed in the SEs associated with CNS Amendment Nos. 183, 187, 196, and 222. As part of the current LAR, the licensee revised the temporal distribution of the $\chi/_Q$ values to conservatively model fumigation of the elevated release to occur during the interval of maximum release to the environment, beginning at 1.3 hours into the event.

3.1.2.2 Radiological Consequences of Design-Basis Accidents

To support the proposed selective implementation of an AST, the licensee analyzed the radiological dose consequences and provided all major inputs and assumptions for the design-basis LOCA.

The information submitted by the licensee reports the results of the radiological consequence analysis for the LOCA to show compliance with dose acceptance criteria expressed in 10 CFR 50.67 for doses offsite and in the control room. The dose acceptance criteria specified in 10 CFR 50.67 provides an alternative to the previous whole body and thyroid dose guidelines stated in 10 CFR 100.11 and GDC 19.

RG 1.183, Regulatory Position 3.1, "Fission Product Inventory," states that

> The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS [emergency core cooling system] evaluation uncertainty. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 or ORIGEN-ARP.

For the LOCA analysis, which postulates substantial core melt, and in accordance with the guidance of RG 1.183, the licensee calculated the core isotopic inventory available for release using the ORIGEN2 isotope generation and depletion computer code, and then multiplied the isotopic specific activities by the relevant power level and release fractions. The NRC staff finds the licensee's use of the cited isotope generation and depletion computer code to be acceptable for establishing the core inventory for AST accident analyses.

As stated in RG 1.183, the release fractions associated with the light-water reactor (LWR) core inventory released into containment for the design-basis LOCA and non-LOCA events have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup of 62,000 megawatt days per metric ton of uranium (MWd/MTU) provided that the maximum linear heat generation rate does not exceed 6.3 kilowatt per foot (kw/ft) peak rod average power for burnups exceeding 54,000 MWd/MTU. The licensee stated in its submittal that CNS conforms to these limits, as it meets these fuel criteria for the current core. Burnup of future core designs in excess of these criteria would invalidate this evaluation and require re-analysis of the associated DBAs, in lieu of a change in the applicable guidance.

To perform independent confirmatory dose calculations for the DBAs, the NRC staff used 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 the resulting radiological consequences at selected receptors.

The following sections discuss the NRC staff's review of the DBA dose assessment performed by the licensee to support its LAR dated October 13, 2008, as supplemented.

3.1.2.2.1 LOCA

The current CNS design-basis LOCA analysis is based on the traditional accident source term described in Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," U.S. Atomic Energy Commission, March 23, 1962 (ADAMS Legacy Library Accession No. 8202010067). The current licensing basis radiological consequence analysis for the postulated LOCA is provided in the CNS USAR Chapter XIV, "Loss-of-Coolant Accident." To support implementation of the AST, the licensee reanalyzed the offsite and control room radiological consequences of the postulated LOCA. This reanalysis was performed to demonstrate that the engineered safety features (ESFs) designed to mitigate the radiological consequences at CNS will remain adequate following implementation of the AST.

The licensee submitted the AST-based reanalysis of the LOCA as an attachment to the LAR. Included in this reanalysis are the assumptions, parameters, and newly calculated offsite and control room doses associated with implementing the AST methodology. The licensee cited RG 1.183 as providing the primary radiological analysis assumptions for their reanalysis of the postulated design-basis LOCA. Specifically, the NRC staff's guidance for analyses of the LOCA is detailed in Appendix A of RG 1.183.

3.1.2.2.1.1 Activity Source

For the LOCA analysis, the licensee assumed that the core isotopic inventory available for release into the containment, is based on maximum full power operation of the core at 2,429 megawatts thermal (MWth), or about 1.0038 times the current licensed thermal power level of 2,419 MWth, in order to account for the ECCS evaluation uncertainty. The NRC staff approved a reduction in ECCS evaluation uncertainty and an increase in rated thermal power to 2,419 MWth, based on a measurement uncertainty recapture analysis, in Amendment No. 231, dated June 30, 2008 (ADAMS Accession No. ML081540280). The burnup and enrichment parameters assumed when determining the core isotopic inventory are within current licensed limits for fuel at CNS. The licensee assumed an end of cycle core average exposure of 33,700 MWd/MTU, equaling approximately 1226 full power days of operation.

The core inventory release fractions and release timing for the gap and early in-vessel release phases of the DBA LOCA were taken from RG 1.183, Tables 1 and 4, respectively. Also consistent with RG 1.183 guidance, the licensee assumed that the speciation of radioactive iodine released from failed fuel is 95 percent aerosol (particulate), 4.85 percent elemental, and 0.15 percent organic. The speciation of radioactive iodine for coolant releases, such as from the ECCS, is 97 percent elemental and 3 percent organic.

3.1.2.2.1.2 Transport Methodology and Assumptions

The licensee calculated the onsite and offsite dose consequences of the design-basis LOCA by modeling the transport of activity released from the core to the environment, while accounting for appropriate activity dilution, holdup, and removal mechanisms. The NRC staff reviewed the licensee's assessment of the following potential post-LOCA activity release pathways:

- Primary Containment (PC) Leakage to Secondary Containment
- PC Leakage Bypassing Secondary Containment
    - Main Steam Isolation Valve (MSIV) Leakage Pathway
    - Other Bypass Pathways
- Engineered Safety Feature (ESF) Leakage

Also, the NRC staff reviewed the licensee's assessment of the following potential post-LOCA shine dose pathways:

- External Activity Plume (Outside Cloud)
- Reactor Building Airborne Activity Cloud
- Core Spray Line
- Activity in the Drywell
- Control Room Emergency Filtration System (CREFS) Filter

The licensee assumed a 2-hour delay in the startup of the ECCS after the onset of gap release, consistent with an assumption of a loss-of-offsite power (LOOP) concurrent with the design-basis LOCA. This assumption was made for the purpose of attributing the onset of the deterministically defined core melt to a specific mechanism in order to remain consistent and conservative with respect to the applicable regulatory guidance and requirements.

For releases into containment, the licensee assumed that activity released from the reactor coolant system is instantaneously and homogeneously well-mixed in the drywell. This assumption is conservative and consistent with the guidance of RG 1.183. Therefore, it is acceptable to the NRC staff. In addition to complying with the regulatory guidance, the licensee takes no credit for postulated ECCS restoration and the resulting thermohydraulic response of cooling water quenching the molten core and core debris in the PC. This hypothetical phenomena is generally assumed to result in the drywell and torus airspace volumes becoming well-mixed for the "light bulb" and torus design of the Mark I containment, like that of CNS, as it is configured with downcomers from the drywell that extend below the surface of the torus suppression pool coolant (wetwell). Also, as a result of the licensee's conservative assumption, no credit is taken for the activity decontamination, or scrubbing, associated with such activity releases into the suppression pool fluid.

By crediting the CNS SLC system capability to introduce sodium pentaborate to act as a buffer into the reactor coolant, the licensee has determined that the suppression pool pH remains above 7 for the duration of the accident. Therefore, in analyzing activity transport from containment, it was unnecessary for the licensee to consider re-evolution of iodine dissolved in the coolant. This analysis of post-LOCA suppression pool pH was reviewed by the NRC as documented below in this SE.

The following subsections detail the NRC staff's review of the licensee's analysis of the post-accident activity release paths and contributors to both control room and offsite dose, as mentioned above.

3.1.2.2.1.2.1 PC Leakage to Secondary Containment

The CNS current design basis containment leak rate (L_a) of 0.635 percent weight per day (percent per day) at containment peak pressure, as reflected in the CNS TS leak rate limit, is assumed in the AST LOCA re-analysis. The design basis leak rate of 0.635 percent per day was reduced to 0.3175 percent per day, 50 percent of the initial value, at 24 hours for the remaining accident duration. This reduction was acceptably justified as conservative by the analogous containment pressure and temperature reductions calculated at that same time step. This pathway was modeled by the licensee as the leakage from the PC that occurs prior to, and after, a sustained negative pressure in the CNS Reactor Building (RB) is established at 5 minutes after the initiation of the LOCA. This 5-minute time is referred to as the drawdown period. The licensee calculated that positive pressure in secondary containment exists for only 210 seconds post-accident, but assumed 5 minutes for conservatism. Prior to drawdown, it is not credible to assume filtration or mixing of activity released into the RB, as it can bypass the filtered pathway and short-circuit the building volume. Therefore, during this drawdown period, the licensee assumed that activity is released directly from the RB at ground level without being filtered or mixed in the available volume. Following drawdown, the licensee assumed the activity was released through the Standby Gas Treatment (SGT) system as an elevated release (after 30 minutes), but again conservatively taking no credit for the RB volume. The licensee also conservatively ignored the postulated 2-minute delay in gap activity release when assessing dose contribution from this pathway.

To maximize release to the environment, the licensee assumed that both SGT system trains are in operation for the first hour, with an assumed failure of a filter heater in one of the trains. The licensee determined this heater failure to be the limiting single failure for determining the AST LOCA radiological consequence. The assumed heater failure results in a reduced SGT system filter efficiency for the train with the failed heater. The licensee stated that an assumed failure of filter heater power in one train requires that the faulted SGT system train be manually secured within 1 hour.

The licensee's model of this release path is conservative and acceptable to the NRC staff.

3.1.2.2.1.2.1.1 Activity Removal in PC by Natural Deposition

The licensee's dose analysis assumed that natural deposition, or sedimentation, of particulate activity occurs in PC, and takes no credit for sprays or other filtration mechanisms that would possibly be available to further reduce the containment activity. The licensee used the accepted simplified natural deposition model from NUREG/CR-6189, referred to as the Powers natural deposition model, as implemented in the RADTRAD dose consequence computer code. The NRC staff generally accepts use of the 10th percentile confidence interval (90 percent probability) natural deposition removal values implemented in the RADTRAD code and used by the licensee. The Powers natural deposition model was derived by correlation to results of Monte Carlo uncertainty analyses of detailed models of aerosol behavior in the containment under accident conditions. The NRC staff agrees that the licensee's model of overall PC activity removal by natural deposition is conservative and, therefore, is acceptable.

3.1.2.2.1.2.2 PC Leakage Bypassing Secondary Containment

This pathway generally characterizes leakage through lines that penetrate the PC and the RB. The licensee postulated that leakage from the PC through penetrations and the closed containment isolation valves in these penetrations would bypass the RB and SGT system filters, thereby resulting in unfiltered releases at ground level. The following subsections discuss the bypass pathways assumed by the licensee.

3.1.2.2.1.2.2.1 MSIV Leakage Pathway

In its design basis analysis, the licensee assumed a total MSIV leakage rate of 300 standard cubic feet per hour (scfh), limited to 150 scfh per main steam line (MSL). However, the licensee has proposed a revision to the CNS TSs that would limit the total allowable MSIV leakage to 212 scfh, with a maximum of 106 scfh per line, when tested at greater than or equal to a pressure of 29 pounds per square inch gauge (psig). Therefore, upon NRC staff acceptance and licensee implementation of this amendment, this limit will become the new licensing basis for CNS, and the analysis performed by the licensee will bound the TS allowable leakage.

For releases through this pathway, the licensee has taken credit for the mitigation of particulate radionuclide and elemental iodine activity. There are a number of mechanisms and processes used to model the mitigation and removal of the activity associated with these radionuclides, such as impaction, sedimentation, and deposition. Although the licensee examined the effect of credible mitigating phenomena taking place in the main steam lines, for conservatism, the licensee only credits activity removal in the main condenser. The following subsection discusses the NRC staff evaluation of this removal credit.

3.1.2.2.1.2.2.1.1 Activity Removal in the Main Condenser

In Amendment Nos. 196 and 206, the NRC staff approved an NPPD evaluation that implemented an Alternate Leakage Treatment (ALT) at CNS. This ALT demonstrated the seismic ruggedness of the Turbine Building, the main condenser, and the MS Pathway, as well as the manual actions needed to configure the pathway. There are no changes to the credited manual actions from those approved by the staff in Amendment No. 206. As documented in Amendment No. 206, the licensee's evaluation of the manual actions required to align the MS Pathway resulted in a 2.5 hour limit for manually aligning 14 of 16 valves, and a time limit of 30 hours for manually aligning the two additional valves and installing the shaft adjustment tools (shaft sealing mechanisms) on the turbine stop valves. The licensee re-evaluated these times using the AST LOCA conditions, and determined that the allowed completion time for alignment of the two additional valves and installation of the turbine stop valve sealing mechanisms would be reduced to 5 hours; the time for alignment of the other 14 valves was unchanged. The licensee stated that alignment of the valves would normally be performed by an on-shift plant operator, and installation of the valve sealing mechanisms would be performed by maintenance personnel deployed out of the Operations Support Center following activation of the emergency response organization. The licensee further stated in its application that, based on walkdowns, it estimated that the time required for a single individual to install the pre-staged turbine stop valve sealing mechanisms would be 30 minutes upon personnel entry into the Turbine Building. At the assumed MSIV leakage rates, airborne activity may take as much as 3 hours to travel through the main steam system and out to the environment through the turbine seals, if leakage is conservatively assumed to be released at that location. The changes do not invalidate the previous basis for the NRC staff's acceptance of the credited manual actions, as documented in the staff's SE for Amendment No. 206, and are therefore acceptable.

The ALT pathway ensures that post-LOCA MSIV activity leakage is transported, via the MSIV and MSLs to the main condenser. The licensee credits removal of airborne activity in the main condenser based upon the methodology presented in the General Electric Boiler Water Reactor Owners Group (BWROG) Topical Report NEDC-31858P-A, Revision 2, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," dated August 1999. Commensurate with this model, the licensee made several conservative assumptions to model the transport of airborne activity through the main steam system and the mitigation of activity release afforded by the main condenser. It its analysis, the licensee conservatively credited only the first condenser shell in which the MSIV leakage enters, and only the condenser free volume above the highest main steam drain line condenser penetration into that shell, for holdup and deposition. The licensee conservatively neglected that the ALT drain pathway condenser penetration enters the main condenser at a much lower elevation, thus reducing the available volume and surface area for holdup and deposition. The licensee's assumption is conservative because the ALT pathway is of larger diameter than the drain line and is, therefore, likely to be a less resistant flow path. In addition, when considering a well-mixed transport of airborne activity, this distinction becomes less significant and all volume and surface areas are potentially available. The licensee also conservatively ignored other deposition surfaces provided by the steel internals of the main condenser.

Consistent with the NEDC-31858P-A methodology, the licensee also developed a model to calculate an effective filter efficiency based on deposition and plateout in the condenser. The use of an effective filter efficiency assumes that the activity transport resulting from the design-basis LOCA is taking place in steady-state. In general, though the steady-state simplification is conservative in many respects (e.g., homogenous core activity, instantaneous mixing), some components of the transport phenomena are less conservatively modeled by this assumption. For large volumes in particular, reaching steady-state takes longer periods of time than for smaller volumes; consequently, using an effective filter efficiency can over-predict removal of activity when compared to the results seen when using a removal rate constant. Therefore, the NRC staff does not explicitly approve the licensee's use of an effective filter efficiency for modeling removal of airborne activity in the condenser. However, due to the overall conservatism in the licensee's model of MSIV leakage, and since the assumption of steady-state is generally conservative, the staff concludes that the credit taken by the licensee for airborne activity mitigation provided by the main steam system is acceptable. Any future changes made to the assumptions used by the licensee in its design-basis LOCA MSIV leakage model are subject to the requirements of 10 CFR 50.59.

The NRC-approved NEDC-31858P-A, Revision 2, topical report implemented by the licensee provides a method to calculate removal of elemental (gaseous) forms of iodine; however, this document does not explicitly calculate removal of particulate forms of iodine and other radionuclides. Therefore, the licensee conservatively treated particulates as elemental iodine, and assumed that they were removed, or deposited, with the same efficiency in the main condenser. The NRC staff agrees that particulates will generally be removed with greater ease than gaseous forms of iodine, as there are more mechanisms to cause this removal that have greater reliability (e.g., gravitational settling). The licensee's treatment is commensurate with the methodology cited in NEDC-31858P-A, Revision 2, as well as reference A-9 of RG 1.183, J. E. Cline, "MSIV Leakage Iodine Transport Analysis," letter report dated March 26, 1991 (ADAMS Accession No. ML003683718).

The NRC staff noted that in Appendix A to Enclosure 1 of the licensee's application, assumption A3.2 states that, "No decay or plate-out of organic iodine within the main condenser is credited." The licensee confirmed that this was a misstatement. The staff notes that, while the licensee takes no credit for plateout of organic iodine in the condenser, it does calculate and credit organic iodine decay. The staff finds it appropriate and acceptable to credit decay of all radionuclides, including organic iodine, in the licensee's design-basis LOCA analysis model.

The NRC staff concluded that the licensee's methodology results in conservative credit for iodine and particulate activity removal in the main steam system and is generally conservative when compared to the calculation of such removal using different models and methodologies, including, but not limited to, the determination of activity removal efficiency based on the Monte

Carlo assessment of aerosol settling velocities described in AEB 98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term." Therefore, because the overall removal credited by the licensee's model is conservative with respect to other previously approved models, the licensee's model is acceptable.

3.1.2.2.1.2.2.2 Other Bypass Leakage Pathways

The licensee examined all potential activity leakage pathways from the drywell directly into the condenser, RB, or environment, and determined that there were no identifiable secondary containment bypass leakage pathways other than the MSIV Leakage Pathway discussed previously.

3.1.2.2.1.3 ESF Leakage

The licensee's model of ESF leakage conservatively assumed that, excluding noble gases, all isotopes that are released to the PC instantaneously transported to, and homogeneously mixed in, the torus water (suppression pool) at the onset of the gap activity release phase. This conservative treatment is consistent with the guidance of RG 1.183. CNS has no TS-prescribed limit on ESF leakage. However, in accordance with Amendment No. 187, the current CNS licensing basis assumes a value of 1,000 cubic centimeters per minute (cc/m), or 0.254 gallons per minute (gpm). For design-basis LOCA conditions, the licensee conservatively increased this assumption to a considerably high leakage rate of 45,000 cc/m (11.89 gpm). This leakage was assumed to begin at the onset of the LOCA and last for the duration of the accident. The licensee calculated the torus water temperature would not exceed 212 degrees Fahrenheit (°F); therefore, consistent with the guidance of RG 1.183, the licensee assumed that 10 percent of the iodine in the leaked ECCS fluid becomes airborne and available for release, while all other particulates remain in the water. This activity is then treated as other PC leakage to secondary containment, as it is released to the environment through the SGT system following drawdown, and taking no credit for holdup or dilution in the RB. Also consistent with the regulatory guidance, the iodine activity was assumed to be 97 percent elemental and 3 percent organic. The licensee's treatment of ESF leakage is conservative and consistent with the guidance of RG 1.183 and, therefore, is acceptable.

3.1.2.3 Direct Shine Dose

The licensee's evaluation of post-LOCA shine doses to control room personnel from the external activity plume, the RB airborne activity cloud, activity in the drywell, a Core Spray system line, and a CREFS filter, and was performed using the MicroShield code. The MicroShield code is point-kernel integration code used for general purpose gamma shielding analyses.

Some of the potentially complex geometries associated with the direct shine dose assessment performed for CNS are generally more effectively modeled using more powerful particle transport codes. Specifically, MicroShield sacrifices accuracy in lieu of simplicity when modeling complex multidimensional systems of sources, shields, and receivers, as point-kernel methods which implement buildup factors can potentially mistreat albedo effects. However, the licensee maintained significant conservatism in its calculation and selection of source composition, shield orientation, and receptor location for its analysis or shine dose. Also, at CNS, the total direct shine dose contribution constitutes a small percentage of the total LOCA dose. Based on its engineering judgment, the NRC staff concluded that the licensee's direct shine dose model implements sufficient conservatism to compensate for potential non-conservative treatment of the modeled geometries by the chosen point-kernel code. Therefore, for the general application of this code as implemented for the design-basis LOCA analysis at CNS, the staff concludes that the licensee's direct shine dose assessment is acceptable.

3.1.2.4 Vital Areas Assessment

In its LAR, the licensee states that the existing TID-14844-based analyses of post-LOCA vital area access, as performed in response to Item II.B.2 of NUREG-0737, are bounding with respect to AST-based analyses. The NRC staff agrees with the licensee's assessment because TID-14844 and its associated whole body and thyroid methodology generally bound the results from analyses based on the AST and TEDE methodology. Additionally, the timing, magnitude, and transport assumptions of the TID-14844 source term make it inherently more severe. Therefore, as described in its submittal, the licensee's existing analyses indicate that CNS will continue to comply with the regulatory requirements for vital areas as given in NUREG-0737 and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The NRC staff concluded that the licensee has sufficiently examined the DBA dose consequences to vital areas and the current analysis bounds the AST analysis. Therefore, the licensee's assessment is acceptable.

3.1.2.5 Conclusions for Radiological Dose Consequences

The licensee concluded that the radiological consequences at the EAB, LPZ, and control room are within the dose criteria specified in 10 CFR 50.67 and the accident-specific dose acceptance criteria specified in SRP 15.0.1 and RG 1.183. These accident-specific dose acceptance criteria for the LOCA are a TEDE of 25 rem at the EAB for any 2 hours, 25 rem at the outer boundary of the LPZ for the duration of the accident, and 5 rem for access to, and occupancy of, the control room for the duration of the accident. The NRC staff finds that the licensee used sufficiently conservative analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 3.1.1 of this SE and with those stated in the design bases of the CNS USAR. The staff also performed independent calculations of the dose consequences of the postulated LOCA releases, using the licensee's assumptions for input to the RADTRAD computer code. The staff's calculations found no deficiencies in the licensee's models or calculations. The major parameters and assumptions used by the licensee, and found acceptable to the staff, are presented in Table 3.2.1 of this SE. The results of the licensee's design-basis radiological consequence calculation are provided in Table 3.2 of this SE. The staff concludes that the EAB, LPZ, and control room doses estimated by the licensee for the LOCA meet the applicable accident dose acceptance criteria and, therefore, are acceptable.

3.1.3 Control Room Habitability and Modeling

To calculate the dose in the control room, the licensee used two air intake flow rates and an assumed unfiltered inleakage rate. One intake flow rate was used to model operation of the

Control Room Air Conditioning System (normal ventilation system) and the other was used to model CREFS. The licensee assumed a normal ventilation intake flow rate of 3235 cubic feet per minute (cfm), a CREFS intake flow of 900 cfm ±10 percent, and an unfiltered inleakage rate of 400 cfm. The licensee states that upon receipt of a LOCA signal, characterized by Reactor Vessel Water Level - Low Low, Level 2 or Drywell Pressure – High, the CREFS is automatically initiated, and achieves isolation within 11 seconds. However, the licensee's analysis conservatively assumes that normal ventilation remains in operation for the first minute of the postulated accident. Therefore, a normal air intake rate plus the assumed unfiltered inleakage is modeled for the first minute of the postulated accident. The unfiltered inleakage assumption bounds the inleakage values reported in the NPPD response to NRC Generic Letter 2003-01, "Control Room Habitability," dated June 12, 2003 (ADAMS Accession No. ML031620248), citing control room inleakage test results. After the first minute, the licensee assumed CREFS to be in operation in for the duration of the accident. The licensee assumed that the CREFS filters provide 99 percent filtration efficiency for particulates, and 90 percent filtration efficiency for elemental and organic forms of iodine, and further reduced these values by 1 percent to account for maximum bypass, consistent with the guidance of RG 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," dated June 2001 (ADAMS Accession No. ML011710176).

When applicable, the licensee used the lower uncertainty bounds for assumed flow rates. The licensee demonstrated the conservatism of this treatment in its May 29, 2009, supplemental letter. The value assumed for unfiltered inleakage into the control room is conservative and provides margin for future measurements of control room inleakage and, therefore, is acceptable. The major parameters and assumptions used by the licensee for modeling the control room are presented in Table 3.2.5 of this SE. The NRC staff reviewed the licensee's control room model as applicable to the calculation of radiological consequences and concluded that it was acceptable.

3.1.4 Conclusions for Section 3.1

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of the postulated DBA analysis with the proposed TS changes. The staff concludes that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0. The staff compared the doses estimated by the licensee to the applicable criteria identified in Section 3.1.1. The staff also concludes with reasonable assurance that the licensee's estimates of the control room, EAB, and LPZ doses will comply with these criteria. The staff further concludes with reasonable assurance that CNS, as modified by this approved 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, analysis assumptions, and input parameters. Therefore, the proposed license amendment is acceptable with respect to the radiological consequences of the LOCA DBA.

3.2 Proposed Changes to the SLC System

3.2.1 Regulatory Evaluation

Section 50.54(o) of 10 CFR Part 50 requires primary reactor containments for water-cooled power reactors to be subject to the requirements of Appendix J to 10 CFR Part 50. Appendix J specifies the leakage test requirements, schedules, and acceptance criteria for tests of the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. Appendix J, Option B, Section III.A., requires that the overall integrated leakage rate must not exceed the allowable leakage rate (L_a) with margin, as specified in the TSs. The overall integrated leakage rate, as specified in the Appendix J definitions, includes the contribution from main steam isolation valve (MSIV) leakage. The licensee is requesting a permanent exemption from Option B, Section III.A., requirements to permit exclusion of the MS Pathway (MSL and the main steam inboard drain line) leakage from the overall integrated leakage rate test measurement.

Appendix J, Option B, Section III.B., requires the sum of the leakage rates of all Type B and Type C local leakage rate tests to be less than the performance criterion (L_a) with margin, as specified in the TS. The licensee is also requesting exemption from this requirement, to permit exclusion of the MS Pathway leakage rates from the sum of the Type B and Type C test leakage rates.

The evaluation of the licensee's request for exemption will be issued by the NRC staff via separate correspondence.

Appendix J, Option B, Section V.B.3., requires that the regulatory guide or other implementation document used by a licensee to develop a performance-based leakage testing program must be included, by general reference, in the plant TSs. CNS TS 5.5.12 requires that leakage rate testing be performed as required by 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by one exception listed in the TSs. This RG endorses, with certain exceptions, Nuclear Energy Institute (NEI) report NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995.

The licensee proposes to revise TS 5.5.12 to add two more exceptions from the guidelines of RG 1.163 and NEI 94-01, which would implement the exemption requested in the licensee's October 13, 2008, application. The licensee also proposes revisions to the leakage rate test acceptance criteria in TS 3.6.1.3 in order to implement the requested exemption.

The NRC staff has previously granted similar exceptions from the requirements of Sections III.A and III.B of Option B, including, for example, Monticello Nuclear Generating Plant, dated December 7, 2006 (ADAMS Accession No. ML062790015), and Browns Ferry Nuclear Plant, Units 2 and 3, dated March 14, 2000 (ADAMS Accession No. ML003691985). The staff considered these precedents in this review.

The NRC staff also used the NRC guideline, “Guidance on the Assessment of a BWR SLC System for pH Control” dated February 12, 2004, to evaluate the SLC system for its ability to perform its AST function of post-LOCA suppression pool pH control (ADAMS Accession No. ML040640364). Several similar applications from boiling-water reactor (BWR) licensees to use the SLC system for the same function as that proposed by NPPD have been approved by the staff (e.g., Edwin I. Hatch Nuclear Plant, Units 1 and 2, Amendment Nos. 256 and 200, respectively, dated August 28, 2008, ADAMS Accession No. ML003691985). The approval of these precedents was based on meeting the guidance criteria found in the above NRC guideline.

3.2.2 <u>Technical Evaluation</u>

3.2.2.1 <u>SLC System</u>

The licensee is proposing to use the SLC system to maintain the pH of the water in the suppression pool at or above 7.0 following a design-basis LOCA with indication of fuel damage. Maintaining the pH of the water above 7.0 following a LOCA ensures that iodine will be retained in the suppression pool water as assumed in the LOCA AST analysis. The licensee proposes to manually initiate the SLC system from the main control room upon detection of symptoms that a LOCA with fuel damage is occurring.

The SLC system is credited for the injection of sufficient sodium pentaborate solution to prevent the re-evolution of iodine from the suppression pool for a 30-day period following a design-basis LOCA.

The NRC staff reviewed the quantity of sodium pentaborate available with respect to the quantity of acid producing debris and radiolytic acid production to confirm adequate pH control, as documented in Section 3.3 below.

The NRC staff reviewed the SLC system with respect to SLC role in delivery of sodium pentaborate to the suppression pool for pH control. The control of pH in the suppression pool is required to mitigate the consequences of a DBA in which fuel is damaged. As such, the new purpose assigned to the SLC is a safety-related function. The licensee stated that the SLC system is classified as non-essential (non-safety related), and is not an ESF system. Therefore, the NRC staff used the guideline, “Guidance on the Assessment of a BWR SLC System for pH Control,” to evaluate the SLC system for its ability to perform its AST function of post-LOCA suppression pool pH control.

The NRC staff reviewed the licensee’s submittal, dated October 13, 2008, Attachment 2, which provided the licensee's analysis of the use of the SLC system for the safety-related function. From its review, the staff concluded that the SLC system is comparable to a system classified as safety-related. As such, the SLC system as designed and installed is a high quality system that provides reasonable assurance that the sodium pentaborate will be injected into the core upon activation, specifically,

(1) The SLC system equipment and piping required for post-LOCA injection of sodium pentaborate solution in the reactor has been designed or qualified to

CNS Seismic Class I requirements in accordance with Appendix A to 10 CFR Part 100 and RG 1.29, "Seismic Design Classification" (August 1973).

(2) The SLC system is required to be operable in the event of an offsite power failure. Therefore, the pumps, valves and controls are powered from standby alternating current (AC) power supply. The pumps and valves are powered and controlled from separate buses and circuits so that a single failure does not prevent system operation.

(3) The applicable components of the SLC system are inspected and tested by the licensee in accordance with the American Society of Mechanical Engineers (ASME) Inservice Inspection and Inservice Testing Programs, as required by 10 CFR 50.55a, "Codes and standards."

(4) The functions of the SLC system are evaluated in the CNS Maintenance Rule program consistent with 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," to provide reasonable assurance that the system will perform reliably.

(5) The post-LOCA mission for the SLC system has been evaluated for environmental qualification of electrical equipment important to safety. Electrical equipment required to operate that is exposed to a harsh environment during its mission time were evaluated and determined to be either qualified or identical to qualified equipment for the SLC post-LOCA mission. All other electrical equipment are exempt from 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," qualifications. Therefore, the SLC system meets the requirements of 10 CFR 50.49 and Appendix A to 10 CFR Part 50.

The licensee stated that applicable plant procedures will be revised and implemented as necessary during the AST implementation phase, so that upon detection of high drywell pressure with high drywell radiation levels associated with the postulated activity release, a manual initiation of SLC injection is executed for a LOCA to maintain suppression pool pH at or above 7.0. The impacted procedures include a station emergency procedure used for LOCA mitigation.

The NRC staff considered components that could be subject to single failure. The licensee stated that the only non-redundant active components of the SLC system are the two check valves in series (one inboard and one outboard) located on containment penetration X-42 for the SLC injection line.

The licensee stated in its submittal that the SLC injection check valves are stainless steel 1½-inch piston lift check valves manufactured by Dresser Valve and Control and are mounted horizontally in the injection line. The SLC injection check valves were procured as original equipment for CNS and were designated as essential equipment. The SLC injection check valves are maintained within the requirements of the CNS 10 CFR Part 50, Appendix B Quality Assurance program and subject to the same requirements as other components installed in safety-related systems. SLC check valve testing is accomplished each refueling outage during the flow test which directs demineralized water into the reactor pressure vessel at rated SLC pump flow, per TS SR 3.1.7.8. The licensee stated that it conducted a review of the CNS maintenance and surveillance history for the SLC system and did not discover any failures of the check valves to open on demand. Similarly, the licensee conducted a review of the industry databases, Equipment Performance and Information Exchange (EPIX) and Nuclear Plant Reliability Data System (NPRDS), and no failures of check valves of this manufacturer and type to open were identified. Although a single failure-to-open of one of the two check valves could prevent SLC injection, the NRC staff concluded that the potential for failure is very low, based on the quality of the component as established by its procurement, periodic testing and inspection, and historical performance. The staff, therefore, concluded that the use of a single penetration of the containment with the identified check valves as described by the licensee is acceptable.

The NRC staff considered the transport of the sodium pentaborate from the reactor vessel to the suppression pool. The SLC system injects the sodium pentaborate into the reactor vessel. The transport of reactor vessel contents including the sodium pentaborate to the pool is by flow through the break (assumed to be a recirculation line break) to the drains that feed the suppression pool. The staff concluded that there would be mixing and transport at some rate and that it was reasonable to assume the concentration of sodium pentaborate in the core would equalize with the concentration in the suppression pool within an acceptable time after SLC injection. As a result, the staff concluded there would be sufficient pH control to prevent iodine re-evolution from the suppression pool.

3.3 Suppression Pool pH

The NRC staff verified the licensee's calculation of pH in the suppression pool water and concluded that the use of sodium pentaborate buffer will maintain a pH higher than 7 for the period of 30 days following a LOCA. Maintaining a basic pH will minimize the amount of radioactive iodine that could be released to the containment from the suppression pool water following an accident.

3.3.1 Regulatory Evaluation

Implementation of the AST by the licensee required re-analyzing several the LOCA DBA using the new source term. Because of improved understanding of the mechanisms of the release of radioactivity, the current accident source term could be replaced by a less restrictive AST. An acceptable accident source term is a permissible amount of radioactive material that could be released to the containment from the damaged core following an accident. The licensee performed its analysis in accordance with the requirements of 10 CFR 50.67. The NRC staff reviewed the licensee's analysis for maintaining the acidity or alkalinity (pH) of the water in the suppression pool at a pH ≥7 (that is, a basic solution as opposed to an acidic solution) for 30 days following a LOCA. In accordance with RG 1.183, maintaining a basic pH will minimize re-evolution of iodine from the suppression pool water following an accident.

3.3.2 Technical Evaluation

After a LOCA, a variety of different chemical species are released from the damaged core. One of them is radioactive iodine. This iodine, when released to the outside environment, will significantly contribute to radiation doses. It is therefore essential to keep the iodine confined within the plant's containment. According to NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," iodine is released from the core in three different chemical forms; at least 95 percent is released in ionic form as cesium iodide (CsI) and the remaining 5 percent as elemental iodine (I_2) and hydriodic acid (HI), with at least 1 percent of each of them. CsI and HI are ionized in water environment and are, therefore, soluble. However, elemental iodine is scarcely soluble. It is of interest, therefore, to maintain as much as possible of the released iodine in ionic form. Unfortunately, in radiation environments existing in containment, some of the ionic iodine dissolved in water is converted into elemental form. The degree of conversion varies significantly with the pH of water. At a higher pH, conversion to elemental form is lower, and at a pH >7, it becomes negligibly small. The relationship between the degree of conversion and pH is specified in Figure 3.1 of NUREG/CR-5950, "Iodine Evolution and pH Control," dated December 1992 (ADAMS Accession No. ML063460464).

The licensee calculated that after a LOCA, the pH of the un-buffered suppression pool will be continuously decreasing due to formation of hydrochloric and nitric acid in containment. Hydrochloric acid is formed from the decomposition of chloride bearing cable insulation by radiation. The licensee's calculation conservatively used a 1.2 multiplier in calculating the inventory of cable material. Summing the contributions from both gamma and beta radiation results in the formation of approximately 4.97E-4 mols/liter of hydrochloric acid during the 30-day period. Nitric acid is formed by irradiation of air and water. Based on the 30-day integrated doses from alpha, beta, and gamma radiation, the licensee calculated in their application that approximately 7.37E-5 mols/liter of nitric acid is formed during the same period. Both nitric and hydrochloric acids are strong acids and will contribute to lowering the pH of the suppression pool water.

As described in RG 1.183, in order to keep iodine dissolved, the suppression pool water should be kept at a pH greater than 7 throughout the 30-day post-LOCA period. In its application, the licensee demonstrated that because of strong acid formation in the containment, this is not achievable without adding buffering chemicals to control the water pH. In order to neutralize the effect of acids, the licensee proposed to inject sodium pentaborate from the SLC system. The main purpose of the SLC system is to control reactivity in the case of control rod failure. However, sodium pentaborate can also act as a buffer. Such buffering action could maintain a basic pH in the suppression pool despite the presence of strong acids. The licensee's analysis assumes initiation of SLC system injection within 6 hours after the event starts, with the full contents of the SLC system tank being injected within 8 hours.

The licensee's calculation assumed that the minimum possible amount of sodium pentaborate is injected. Based on the CNS TS Section 3.1.7, the minimum allowable inventory of sodium pentaborate decahydrate is 4,444 pounds. The licensee's analysis assumes that 4,444 pounds of sodium pentaborate decahydrate is injected by the SLC system. Using the minimum amount of sodium pentaborate is a conservative input to the analysis because it will result in a lower calculated pH rather than using a larger quantity of sodium pentaborate.

In order to evaluate the beneficial effect of the sodium pentaborate, the licensee calculated suppression pool pH for unbuffered and buffered cases. As expected, without addition of sodium pentaborate, but taking only credit for the presence of Cs(OH), the value of pH during the 30-day period was below 7, reaching a minimum pH value of 3.32. However, with the addition of sodium pentaborate, the pH will increase rapidly above 7 and remains above a pH of 8 for the 30 days post-LOCA.

The NRC staff independently verified the licensee's calculations and finds that by using sodium pentaborate as a buffer, the pH of the suppression pool will remain above a pH of 7 for 30 days post-LOCA.

3.3.3 Conclusion for Section 3.3

The NRC staff reviewed the licensee's assumptions to minimize iodine re-evolution as presented in the re-analysis of the radiological consequences for a LOCA. The methodology relies on using buffering action of sodium pentaborate, introduced into the suppression pool from the SLC system. The assumptions are appropriate and consistent with the methods accepted by the staff for the calculation of post-accident containment sump pH. In addition, the staff independently verified that the post-accident containment sump pH will be maintained above 7 for 30 days following a LOCA. Since the licensee's analysis is consistent with the guidance of RG 1.183, the NRC staff concludes the proposed changes are acceptable.

3.4 Seismic Qualification of Components

The NRC staff reviewed the seismic qualification of components that relate to the transport and removal mechanisms related to the source term. Specifically, the seismic qualification of portions of the licensee's MS Pathway, SLC system equipment, and heating, ventilation, and air conditioning (HVAC) system components (including the CREFS components) which were credited in the licensee's application for an AST were reviewed to determine whether they would maintain continued safe operation under design-basis seismic loading conditions following the AST implementation.

3.4.1 Regulatory Evaluation

Pursuant to 10 CFR 50.67, a licensee may revise its current accident source term by re-evaluating the consequences of DBAs with the AST. The NRC's regulatory guidance associated with the implementation of an AST is provided in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." NPPD is requesting the approval of a selective-scope AST for the design-basis LOCA, as described in RG 1.183.

In its review, the NRC staff also used the guidance found in NUREG-0800, SRP Section 15.0.1. The NRC has recently issued similar AST implementation license amendments for BWRs at Peach Bottom Atomic Power Station, Units 2 and 3 (Amendment Nos. 269 and 273, respectively, dated September 5, 2008; ADAMS Accession No. ML082320406), Nine Mile Point Nuclear Station, Unit 1 (Amendment No. 194, dated December 19, 2007; ADAMS Accession

No. ML073230597), and Limerick Generating Station, Units 1 and 2 (Amendment Nos. 185 and 146, respectively, dated August 23, 2006; ADAMS Accession No. ML062210214).

Additional regulatory guidance regarding topics specific to the mechanical and civil engineering review can be found in the NEDC-31858P-A, Revision 2, "BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems." The NRC staff's SE for the topical report documents the NRC staff's approval and provides additional guidance for licensees regarding the topic of MSIV leakage pathways.

3.4.2 Technical Evaluation

3.4.2.1 MS Pathway Evaluation

In performing its re-evaluation of the design-basis LOCA to support implementation of the AST, the licensee stated in its application that it would take credit for the reduction of the amount of radioactivity released through MSIV leakage by deposition and plateout in the main condenser at CNS. While the licensee addressed the entire Alternate Leakage Treatment (ALT) pathway (defined as the "Main Steam Pathway" (MS Pathway) at CNS), the licensee conservatively neglected portions of the MS Pathway other than the condenser in its AST analysis. With regards to the accreditation of components used for the deposition and plateout of MSIV leakage, Position 6.5 of Appendix A of RG 1.183 (LOCAs) states, in part, that:

> A reduction in MSIV releases that is due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, may be credited if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake (SSE).

In its submittal of the CNS seismic evaluation dated February 26, 2002 (ADAMS Accession No. ML020650643), the licensee described its use of the NRC-approved methodology for demonstrating the seismic ruggedness of the turbine building, main condenser, and piping systems making up the MS Pathway. In Amendment No. 196, dated February 21, 2003 (ADAMS Accession No. ML030560804), the NRC staff approved the licensee's proposed methodology for evaluating the seismic adequacy of the MS piping, the main condenser, and the turbine building.

The proposed methodology is described in NEDC-31858P-A, Revision 2, and was subsequently approved by the NRC staff in its SE dated March 3, 1999 (ADAMS Accession No. ML010640286). According to Sections 4.5.2, 4.5.3, and 4.5.4 (corresponding to the turbine building, main condenser, and MS Pathway piping, respectively) of the licensee's February 26, 2002, letter, in accordance with NEDC-31858P-A, Revision 2, the licensee utilized walkdowns, detailed analyses, and earthquake experience data to seismically qualify the components of the MS Pathway at CNS. More specifically, the condenser design data for CNS was determined to be bounded by seismic design data for similar sites which had exhibited acceptable earthquake performance. In Section 4.5.3.3 of the licensee's February 26, 2002, letter, the anchorages for the condenser at CNS were analytically determined to be acceptable for design-basis seismic events. In Amendment No. 196, the NRC staff concluded that the methodology utilized by the licensee for determining the seismic adequacy of the MS Pathway at CNS was acceptable.

By application dated December 9, 2003 (ADAMS Accession No. ML033490541), the licensee requested approval of the final configuration of the CNS MSIV leakage pathway concurrent with a request for permanent use of a LOCA dose calculation methodology. In Section 3.1.3 of the SE for Amendment No. 206, dated September 1, 2004 (ADAMS Accession No. ML042470174), the NRC staff concluded that the licensee's MSIV leakage pathway to the main turbine condenser was acceptable based on the methodologies referenced in topical report NEDC-31858P-A, Revision 2.

Based on the NRC staff's previous SEs for Amendment Nos. 162 and 206, the staff concludes that the licensee's MS Pathway, which is being credited for the proposed LOCA AST, is seismically adequate and will maintain its structural integrity under design-basis conditions.

3.4.2.2 SLC System Seismic Evaluation

In performing the re-evaluation of the LOCA DBA, the licensee stated in Attachment 1 of its submittal dated October 13, 2008, that credit would be taken for controlling the pH in the suppression pool following a LOCA by injecting sodium pentaborate into the reactor core using the standby liquid control (SLC) system. In "Guidance on the Assessment of a BWR SLC System for pH Control" dated February 12, 2004, the NRC staff provided guidance on demonstrating that the SLC system was capable of performing its intended safety function during a LOCA following AST implementation. This guidance was developed by the NRC staff during its evaluation approving the AST for Edwin I. Hatch Nuclear Plant, Units 1 and 2 (Amendment Nos. 256 and 200, respectively, dated August 28, 2008). In its application, the licensee addressed the above-referenced NRC staff guidance to demonstrate that the SLC system is credited as either safety-related or comparable to a safety-related system. The licensee indicated that the SLC system at CNS was classified as non-safety related and provided its justification for using the SLC system for safety-related purposes.

One of the key criteria in demonstrating the classification of the SLC system as a safety-related system is the system's seismic qualification, which the licensee addressed in Attachment 2 of its October 13, 2008, letter. The licensee stated that the system equipment and piping for the SLC system at CNS was designed and qualified in accordance with the seismic design methodologies described in the CNS USAR. The licensee also described the redundancy of the active components within the SLC system, in accordance with the guidance in "Guidance on the Assessment of a BWR SLC System for pH Control." This fourth criterion in the guidance recommended that licensees address any non-redundant, active components in detail using one of three response options. The licensee stated in Attachment 2 of its October 13, 2008, letter that the only active non-redundant components within the SLC system are two check valves within the system: CNS-3-SLC-CV-12V and CNS-3-SLC-CV-13CV. In addressing the non-redundant nature of these valves, the licensee provided the information regarding the design-basis conditions under which these valves may operate, including environmental and seismic conditions. The licensee stated that the valves were essential components which were designed to Class I seismic requirements, thus satisfying the information requested by the guidance in "Guidance on the Assessment of a BWR SLC System for pH Control" dated February 12, 2004.

Based on the information provided by the licensee, the aforementioned components were designed and qualified to Class I seismic requirements. Therefore, the NRC staff concurs with the licensee's assessment that the aforementioned system equipment will continue to operate safely upon implementation of the proposed AST.

3.4.2.3 HVAC Ductwork Evaluation

The licensee indicated in Section 4.1.7 of Attachment 1 in its October 13, 2008, letter that credit would be taken for portions of the Control Room HVAC and CREFS systems for the purpose of dose reduction in support of the proposed AST. The licensee provided supplemental information in its April 8, 2009, letter regarding the seismic qualification of the systems and to clarify the accreditation of these systems for the proposed AST. As indicated in Section 4.1.7 in Attachment 1 of the licensee's October 13, 2008, submittal, the Control Room HVAC system is assumed to operate for 1 minute upon receipt of a LOCA signal while the CREFS system is assumed to begin operation approximately 11 seconds following the same signal. Based on this operational sequence, no credit was taken for the dose reduction by the Control Room HVAC system. However, in its April 8, 2009, letter, the licensee indicated that the ductwork and HVAC Supply Fans (HV-FAN-(SF-C-1A & 1B)) are classified as essential and Seismic Class I components. The licensee also stated in its April 8, 2009, letter, that the following components of the CREFS system are classified as essential and Seismic Class I: Control Room Emergency Bypass Pre-Filter (HV-F-(PF-C-IA)), the High Efficiency Filter (HV-F-(HEF-C-1 A)), the Carbon Filter for Control Room Emergency Bypass (HV-F-(CF-C-I A)), the Control Room Emergency Supply Fan (HV-FAN-(BF-C-IA)), and the Emergency Supply Fan Motor (HV-MOT-(BF-C-I A). Since these components are classified as essential and Seismic Class 1, the licensee stated in its April 8, 2009, letter that these components are structurally qualified to operate under design-basis seismic conditions as described in the CNS USAR, Section 2.3.5.1.1.

Based on the above, since the ductwork and components of the Control Room HVAC system and the CREFS system would be able to perform their dose reduction functions under a design-basis seismic event, the NRC staff concurs with the licensee's assessment that these systems will continue to operate safely upon implementation of the proposed AST.

3.4.2.4 Conclusions for Section 3.4

The NRC staff has reviewed the licensee's assessment of the impact of the proposed changes associated with the implementation of the selective scope AST methodology at CNS on portions of the MS Pathway, SLC system, and HVAC system components with regard to the seismic qualification involved with these components as they relate to the AST implementation. Based on the above, the NRC staff concludes that the proposed AST implementation will not have an adverse impact on the ability of these systems to withstand and perform their intended safety functions when subjected to a design-basis seismic event. Therefore, the staff concludes that the proposed changes are acceptable.

3.5 Consideration of Boron Precipitation

Implementation of AST for LOCA in BWRs involves use of the SLC system to control the pH level in the suppression pool during mitigation of a LOCA. As a result, the licensee proposes to revise the CNS TS for the SLC system. In BWRs, the SLC system was designed to mitigate an anticipated transient without scram (ATWS) event. Boric acid solution is stored in the SLC tank and injected inside the lower plenum of the reactor pressure vessel as a means to shut down the reactor following an ATWS. The SLC system was not originally intended to be used during a LOCA.

The NRC staff evaluated whether the use of the boric acid solution from the SLC system following the design-basis LOCA could result in boron precipitation in the core during the long-term cooling phase, and thereby degrade core cooling during the LOCA.

3.5.1 Regulatory Evaluation

The licensee proposes to implement the AST for the design-basis LOCA based on the guidance provided in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." The NRC's acceptance criteria for the design-basis LOCA are based on: (1) 10 CFR 50.46, which establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) 10 CFR Part 50, Appendix K, which establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a LOCA; and (3) 10 CFR Part 50, Appendix A, GDC-35, "Emergency core cooling," which requires, in part, that

> A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented...

3.5.2 Technical Evaluation

As stated in NUREG-1465, the iodine entering the containment from the reactor coolant system during an accident would be composed of at least 95 percent cesium iodide (CsI). Upon deposition on interior surfaces and dissolution in the suppression pool, the predominant form of the iodine would be the iodide ion (I^-). At a pH less than 7.0, a large fraction of the iodide could be converted by irradiation into elemental (gaseous) iodine (I_2) and released into the containment atmosphere. However, if the pH is maintained above 7.0, the fraction of I^- converted into I_2 is expected to be less than 1 percent.

One way to minimize the release of gaseous iodine is to add an alkaline chemical capable of buffering the pH at a value above 7.0. The licensee proposes to do this at CNS by adding sodium pentaborate from the SLC system following a LOCA. Although the SLC system was designed as a backup method to maintain the reactor subcritical without control rods after an ATWS, it can be used for pH control. The licensee proposes to use the SLC system to inject sodium pentaborate into the lower plenum of the reactor pressure vessel, where it will mix with

ECCS flow and spill over to the drywell and then to the suppression pool. Sodium pentaborate is a base, and will neutralize acids generated in the post-accident PC environment.

The licensee stated that a combination of known parameters and conservative assumptions were used as inputs to calculate pH at discrete times for 30 days following the postulated accident. Credit for the SLC system in the radiological analyses is based on operation of one SLC pump, initiated within 6 hours after the event starts, with injection completed within 8 hours. This credit assumes the injection of the entire contents of the SLC system sodium pentaborate solution storage tank with a solution concentration that meets the limits specified in CNS TS 31.7. As an implementing action following issuance of the amendment, the licensee stated in its application that CNS operating procedures will be revised to direct operators to manually initiate the SLC system upon detection of symptoms indicating that a LOCA with core damage is occurring.

The licensee stated that these changes do not require any physical modification of the plant, and do not result in any change to normal plant operation. The licensee also stated that this additional use of the SLC system does not compromise or adversely affect the function of the SLC system as a separate means from the control rods of shutting down the reactor.

In order to evaluate whether it is likely for boron injected from the SLC system to precipitate in the core, the NRC staff requested the licensee to provide following additional information:

i) Describe the flow path of boron solution mixture after initiation of the SLC system following a LOCA, including the path through the core.

ii) Justify how sustained and continuous boil-off of water mixed with the boron solution inside the core will not cause boron concentration to increase with time, and eventually precipitate in the core during the residual heat removal (RHR) cooling mode of a LOCA.

iii) Describe what measures are taken to monitor boron concentration in the core after the SLC system is initiated following a LOCA, and how to prevent potential boron precipitation in the core due to sustained boil-off in the core.

In response to the NRC staff's questions, the licensee stated in its supplemental letter dated April 8, 2009, that the sodium pentaborate solution from the SLC tank is piped into the reactor vessel and is discharged near the bottom of the core shroud so it mixes with the cooling water rising through the core. The boron solution will mix with the ECCS flow and spill over to the drywell through the break and then to the suppression pool.

In the event of a DBA LOOP concurrent with a LOCA (DBA-LOOP-LOCA), two RHR pumps and one core spray pump will initially be available to circulate at least 18,050 gpm of water from the suppression pool to the reactor vessel. Based on the maximum pool inventory of 103,979 cubic feet, this ECCS flow represents one complete exchange of the suppression pool volume every 43 minutes. After 600 seconds, it is assumed that one RHR pump will be reconfigured to suppression pool cooling. The suppression pool is assumed to be well-mixed such that a single pool pH value can be applied. After injection, a fixed quantity of boron and water continue to be circulated in a closed loop flow path.

The injection of the flow from the RHR pump and the core spray pump will also keep the sodium pentaborate solution in the reactor vessel well mixed, both during the initial injection and post-injection periods. The total ECCS flow is substantially higher than the SLC injection rate of 38.2 gpm per pump (76.4 gpm total), and will be well mixed before it overflows to the drywell and then to the suppression pool. Once initiated, the contents of the SLC tank (up to 4,416 gallons) will be injected in less than 2 hours, assuming only one pump in operation. During the time when sodium pentaborate solution is being injected, one to two complete exchanges of the pool volume are expected to occur. This will result in a steady increase in the boron concentration in the pool during the injection process.

The licensee stated that the SLC system was designed to provide a minimum of 660 parts per million (ppm) boron in the reactor vessel after injection. To ensure this minimum concentration is achieved, the SLC volume was determined based on a target value of 125 percent, or 825 ppm. The total weight of water in the reactor (including the recirculation loops) is 706,000 pounds, and the RHR system is an additional 256,000 pounds, for a total of 962,000 pounds. This is equivalent to approximately 15,506 cubic feet of water. The volume of the suppression pool is estimated to be 103,979 cubic feet. When these two volumes are combined after a LOCA, the total volume is approximately 119,485 cubic feet. This will provide a dilution factor of about 7.7, and will result in a final boron concentration of about 110 ppm in the water being circulated by the ECCS system. This is significantly below the saturation concentration for sodium pentaborate solution. At a temperature of zero degrees centigrade, the saturation concentration of sodium pentaborate in solution is 8 percent.

Based on (1) the low concentration of boron in suppression pool and reactor vessel, (2) the short period during which boron injection will take place, (3) the solution will be well mixed at the start of injection and will remain well mixed during the injection due to high volume of ECCS flow between suppression pool and reactor vessel, and (4) the mixing due to natural circulation in the core and lower plenum region, boron precipitation is not considered to be a concern in BWRs.

The licensee further stated that post-accident sampling during a LOCA will be performed using the Post-Accident Sampling System, which permits sampling the reactor coolant and the RHR system water. Samples are analyzed for pH level, and may also be analyzed for boron. Since the water will be well-mixed, no additional sampling in the core region is necessary.

After reviewing the information provided by the licensee, as discussed above, the NRC staff concludes that since the rates at which ECCS water is injected by core spray (at the top of the core) and by RHR pumps (at the lower plenum of the vessel) are substantially higher than the core boil-off rate, the boron solution is not expected to remain stagnant inside the core region as the boil-off occurs. Instead, the injected solution should flow out of the core through the core inlet and mix with the rest of the water in the core. This should prevent the boron concentration from rising significantly inside the core due to sustained boil-off. The colder water sprayed at the top of the core by core spray should help keep the boron solution mixed inside the core by the natural circulation process. In addition, since the boron solution remains very diluted and well-mixed throughout the period, it is unlikely that the boron concentration can rise to a level that could cause boron precipitation inside the core any time during the long-term cooling phase. Therefore, the NRC staff has reasonable assurance that boron precipitation and the resulting degradation of core cooling will not occur during the long-term cooling phase of a design-basis LOCA with SLC system injection in a BWR.

3.5.3 Conclusions for Section 3.5

The NRC staff evaluated the proposed changes to determine whether adequate core cooling is maintained following a design-basis LOCA with SLC system injection, and that the applicable regulations and requirements continue to be met. Since the staff has reasonable assurance that boron precipitation and the resulting degradation of core cooling will not occur during the long-term cooling phase of a design-basis LOCA with SLC system injection, the staff concluded that applicable regulatory requirements will continue to be met. Therefore, the proposed changes are acceptable.

3.6 Electrical Engineering Review

The NRC staff reviewed the proposed changes with regard to the impact on (1) environmental qualification of affected equipment and (2) the electrical systems.

3.6.1 Regulatory Evaluation

The following NRC requirements and guidance documents are applicable to the staff's review.

10 CFR Part 50, Appendix A, General Design Criterion (GDC) 17, "Electric power systems," requires, in part, that nuclear power plants have onsite and offsite electric power systems to permit the functioning of structures, systems, and components that are important to safety. The onsite system is required to have sufficient independence, redundancy, and testability to perform its safety function, assuming a single failure. The offsite power system is required to be supplied by two physically independent circuits that are designed and located so as to minimize, to the extent practical, the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. In addition, this criterion requires provisions to minimize the probability of losing electric power from the remaining electric power supplies as a result of loss of power from the unit, the offsite transmission network, or the onsite power supplies.

GDC 18, "Inspection and testing of electric power systems," requires that electric power systems that are important to safety must be designed to permit appropriate periodic inspection and testing.

10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," requires that the safety related electrical equipment which are relied upon to remain functional during and following design basis events be qualified for accident (harsh) environment. This provides assurance that the equipment needed in the event of an accident will perform its intended function.

10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," requires that preventative maintenance activities must not reduce the overall availability of the systems, structures, or components.

Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," states that the licensees may use either the AST or the Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," assumptions for performing the required environmental qualification (EQ) analyses to show that the equipment remains bounding. RG 1.183 further states that no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST versus TID 14844) on EQ doses.

RG 1.75, Revision 3, "Criteria for Independence of Electrical Safety Systems," describes a method acceptable to the NRC staff for complying with the NRC's regulations with respect to the physical independence requirements of the circuits and electric equipment that comprise or are associated with safety systems.

3.6.2 Technical Evaluation

The licensee proposes to use an AST to determine the offsite and control room doses resulting from a LOCA. The licensee proposes to use the Standby Liquid Control (SLC) System to control the pH of the suppression pool during mitigation of a LOCA. Maintaining the suppression pool pH above 7.0 will limit the evolution of gaseous iodine released into the containment. The licensee stated in its application that no plant modifications are planned to implement the LOCA AST analysis.

The NRC staff requested additional information on changes to the loading sequence of the CNS emergency diesel generators (EDGs) to support the license amendment request. The licensee stated in its September 1, 2009, supplemental letter that no loads were added to the CNS EDGs as a result of the AST adoption and that the SLC system has always been connected to the emergency buses. Thus, the loading sequences of the EDGs are unaffected by this license amendment request.

The staff further questioned whether any loads were being added to the CNS EDGs and, if so, how the loads being added would affect the capability and capacity of the EDGs. The licensee stated in its September 1, 2009, letter that no loads were added to the CNS EDGs as a result of the AST adoption. The SLC system is manually initiated and can be initiated up to 6 hours after a LOCA. The licensee further stated that operators can add additional load to the EDGs provided that the EDGs do not exceed design load limitations. The CNS Updated Safety Analysis Report (USAR) states that the continuous rated capacity of the EDGs is 4000 kilowatts (kW) and has an overload capacity of 4400 kW for 2 hours per day, 5000 kW for 320 hours total, or 4700 kW for 2000 hours per year. The EDG loading tables in the USAR indicate that for 0.5-7 days, the loading is 3983 kW for EDG No. 1 and 3836 kW for EDG No. 2, both of which are under the 4000 kW continuous rating. Per the licensee's operating procedures, the operators can shed non-essential loads on the EDGs to maintain the load under 4000 kW, in order to add the SLC pump (44.5 kW). Thus, the EDGs can support the manual initiation of the SLC system.

Since the SLC system is a non-safety related system, the staff requested additional information regarding how the SLC system will be electrically separated from the safety-related system. Specifically, the staff requested information on how a fault on the non-Class 1E electrical circuit will not propagate to the Class 1E circuit. The USAR states that two sets of the components, SLC pumps and explosive valves, are provided in parallel redundancy. The licensee stated in its September 1, 2009, letter that separate essential motor control centers supply each of the two SLC system pumps, the squib valves, and associated continuity meters. The EDGs provide standby power to the essential buses. Thus, each division is fed from separate essential power. Furthermore, the licensee stated that there is no potential for a fault on a non-Class 1E circuit to propagate to a Class 1E circuit. In the case that the SLC heaters fail, essential power is still available to the SLC pumps and squib valves. Based on the above information, the NRC staff finds that the SLC system has sufficient redundancy and independence.

In addition, the staff requested the licensee to describe how the SLC meets the single failure criterion. The CNS USAR states that the pumps and valves are powered and controlled from separate buses and circuits so that a single electrical failure will not prevent system operation. The licensee stated in its letter dated September 1, 2009, that since the CNS design was developed during a similar timeframe as the regulations and standards regarding single failure, CNS is only required to meet the single failure criteria for those items to which the licensee has committed. The SLC system is not required to meet the single failure criteria. However, the SLC system consists of two independent pumps and squib valves, as discussed above. Therefore, there is redundancy in the SLC system to assure that the safety objectives of maintaining subcriticality are met. The NRC staff concludes that this meets the intent of the single failure criterion.

The staff requested additional information on how the operators would be notified in the event that the SLC may become inoperable. In its letter dated September 1, 2009, the licensee stated that the control room has the following alarms: SLC tank hi/low level, SLC tank hi/low temperature, loss of continuity to the squib valves and SLC tank heater ground to solution. Furthermore, the licensee stated that to initiate the SLC system, the operator turns a key-locked switch and verifies the pump starts by observing pump discharge pressure and indicating lights. Based on the above, the NRC staff concludes that there is adequate indication to operators in the control room to reveal when the SLC may be inoperable.

The staff requested the licensee provide a list and descriptions of components added to its 10 CFR 50.49 program due to the AST and, additionally, confirm that these components are qualified for the environmental conditions to which they are expected to be exposed. In its September 1, 2009, letter, the licensee stated that no components were added to the CNS 10 CFR 50.49 program as a result of the AST adoption. For components outside the drywell, the operating environment remains unchanged. However, due to the chemical spray that would be initiated in containment as a result of SLC injection, the licensee stated that it has updated the EQ profiles and will implement the updated profiles as part of implementation of this license amendment. The licensee provided a table of the affected equipment types and stated that the EQ equipment will remain qualified for the spray conditions that will exist if SLC is initiated following a LOCA. The licensee stated that one of the following approaches was used to show that the equipment remains qualified:

1. Credit the chemical spray composition in existing Institute of Electrical and Electronics Engineers (IEEE) qualification test program(s).

2. Use separate effects testing or analysis to demonstrate material compatibility with Sodium Pentaborate.

3. Device is hermetically sealed or otherwise protected from chemical spray.

4. Component is located in EQ Zone PC1, which is above the highest spray header elevation.

5. Demonstration that the component has performed its function prior to manual spray initiation.

Based on the above, the NRC staff concludes that the equipment in the drywell and containment is qualified for the environmental conditions to which it is expected to be exposed.

The NRC staff also reviewed the EQ portion of the license amendment request. The licensee stated in its application that it used the methodology contained in TID 14844 to determine the radiation doses in the existing EQ analyses. As mentioned previously, the use of this methodology is consistent with the guidance contained in RG 1.183. Since the licensee will continue to use the TID 14844 methodology, and no new equipment is added to its 10 CFR 50.49 program, the EQ analyses of affected equipment should bound the implementation of the AST for LOCA. Therefore, the proposed changes are acceptable.

Based on the above, the NRC staff concludes that the proposed changes comply with the regulatory requirements listed above in section 3.6.1 and are consistent with the guidance in RGs 1.183 and 1.75. Therefore, the proposed changes are acceptable.

3.7 <u>Proposed TS Changes</u>

1. The licensee proposes to revise TS 1.1, "Definitions," to change the definition of DOSE EQUIVALENT I-131 to reflect the dose conversion factors contained in Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration," Environmental Protection Agency, 1988 (second printing with corrections, 1989). This change is proposed pursuant to the guidance contained in RG 1.183, Regulatory Position 4.1.2.

   With the implementation of AST, the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, GDC 19, are replaced by the TEDE criteria of 10 CFR 50.67(b)(2). This new definition reflects adoption of the dose conversion factors and dose consequences of the revised radiological analyses. Thus, this proposed revision to the definition of DOSE EQUIVALENT I-131 is supported by the justification for the proposed licensing basis revision to implement the AST, and conforms to the implementation of the AST and the TEDE criteria in 10 CFR 50.67. Therefore, the proposed change is acceptable.

2. The licensee proposes to revise TS 3.1.7, "Standby Liquid Control (SLC) System," by adding MODE 3 to the APPLICABILITY section, and adding Required Action C.2 to be in MODE 4 within 36 hours. These changes are needed since a LOCA could occur in MODES 1, 2, and 3. These TS changes make the SLC system consistent with other structures, systems, and components contained in the CNS TSs for LOCA mitigation, and are therefore acceptable.

3. The licensee proposes to revise TS SR 3.6.1.3.10 by replacing the combined leakage rate with a limit for each MSIV line. The new allowable MSIV leakage rate for each line would be "≤ 106 scfh when tested at ≥ 29 psig."

   New SR 3.6.1.3.12 is proposed that would establish a new allowable aggregate Main Steam Pathway leakage limit, increasing the current SR 3.6.1.3.10 limit of "≤ 46 scfh when tested at ≥ 29 psig" to "≤ 212 scfh when tested at ≥ 29 psig." Maintaining leakage within these values ensures that the analyzed dose contribution via this pathway remains bounding. This is consistent with assumptions included in the design-basis LOCA radiological consequence analysis. The NRC staff performed confirmatory calculations to verify that the proposed SRs are bounded by the licensee's analyses. In addition, the changes are consistent with the licensee's proposed exemption from 10 CFR Part 50, Appendix J.

   The inclusion of leakage limits (≤ 106 scfh for each MSIV line and ≤ 212 scfh for the MS Pathway) at ≥ 29 psig serves to identify leakage requirements at a reduced test pressure (the MSIVs are tested at the reduced pressure, and not at Pa). The reduced test pressure leakage rates of ≤ 106 scfh and ≤ 212 scfh were determined based on the methodology as designated in the American Society of Mechanical Engineers (ASME) Operating and Maintenance Code, Section ISTC-3600. Therefore, the proposed changes are acceptable.

4. The licensee proposes to revise TS 5.5.12.a.4 to reflect an exemption from Section III.A of 10 CFR 50, Appendix J, Option B, to allow the leakage contribution from the MS Pathway (Main Steam lines and the Main Steam inboard drain line) leakage to be excluded from the overall integrated leakage rate from Type A tests. CNS currently has an approved exemption for the MSIV leakage. This change adds the leakage through the MS inboard drain line. Based on approval of the exemption, the proposed change is acceptable.

5. The licensee proposes to revise TS 5.5.12.a.5 to reflect an exemption from Section III.B of 10 CFR 50, Appendix J, Option B, to allow the contribution from the MS Pathway (Main Steam lines and the Main Steam inboard drain line) leakage to be excluded from the sum of the leakage rates from Type B and Type C tests. CNS currently has an approved exemption for the MSIV leakage. This change adds the leakage through the MS inboard drain line. Based on approval of the exemption, the proposed change is acceptable.

4.0 STATE CONSULTATION

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

5.0 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. 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 published in the *Federal Register* on January 23, 2009 (74 FR 4251). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

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Date: September 15, 2009

Table 3.1.1

CNS LOCA SGT System Elevated Release
Atmospheric Dispersion Factors ($\chi/_Q$ Values, sec/m^3)

Period (hours)	EAB	LPZ	Control Room
0 - 0.083	1.6×10^{-5}	4.00×10^{-5}	4.15×10^{-3}
0.083 – 1.3	1.6×10^{-5}	4.00×10^{-5}	1.00×10^{-10}
1.3 – 1.8	1.2×10^{-4}	1.40×10^{-4}	3.03×10^{-4}
1.8 – 2.0	1.6×10^{-5}	4.00×10^{-5}	1.00×10^{-10}
2.0 – 8.0	-----	4.00×10^{-5}	8.58×10^{-10}
8.0 – 24	-----	1.60×10^{-5}	1.41×10^{-8}
24 – 96	-----	5.80×10^{-6}	5.62×10^{-9}
96 - 720	-----	1.70×10^{-6}	5.69×10^{-9}

Table 3.1.2

CNS LOCA Turbine Building Ground Level Diffuse Release
Atmospheric Dispersion Factors ($\chi/_Q$ Values, sec/m^3)

Period (hours)	EAB	LPZ	Control Room
0.0 – 2.0	5.20 x 10-4	2.90 x 10-4	8.64 x 10-4
2.0 – 8.0	-----	2.90 x 10-4	4.66 x 10-4
8.0 -24	-----	7.30 x 10-5	2.32 x 10-4
24 - 96	-----	2.50 x 10-5	1.53 x 10-4
96 - 720	-----	5.20 x 10-6	1.25 x 10-4

Table 3.2

Licensee Calculated Radiological Consequences of Design Basis LOCA at CNS

Design Basis Accident	Control Room		[1]EAB		LPZ	
	[2]Total Dose (rem TEDE)	Acceptance Criteria (rem TEDE)	[3]Total Dose (rem TEDE)	Acceptance Criteria (rem TEDE)	[4]Total Dose (rem TEDE)	Acceptance Criteria (rem TEDE)
LOCA	2.88E+00	*5.0*	1.00E+00	*25*	5.60E+00	*25*

[1] The licensee calculated the EAB dose for the worst 2-hour period of the accident duration.
[2] The licensee's control room dose results have been rounded to three significant digit precision.
[3] The licensee's EAB dose results have been rounded to three significant digit precision.
[4] The licensee's LPZ dose results have been rounded to three significant digit precision.

Table 3.2.1

Key Parameters Used in Radiological Consequence Analysis of Loss of Coolant Accident

Parameter	Value
Reactor Core Power, MWth	2429
Primary Containment Volume, ft^3 Drywell Airspace Suppression Pool	 132,250 96,445
Secondary Containment Volume, ft^3	200 (used to model no mixing or dilution)
Drywell and Wetwell Airspace Mixing Initiation, hrs	None
Primary Containment Leakage Rate, weight % per day 0 to 24 hrs 24 hrs to 30 days	 0.635 0.3175
MSIV Leakage Rate, scfh Per MSL Total	 150 300
Leakage Activity Removal, % Particulate Effective Filter Efficiency 0 – 24 hours 24 – 720 hours Elemental Iodine Effective Filter Efficiency 0 – 24 hours 24 – 720 hours Organic Iodine Effective Filter Efficiency 0 – 24 hours 24 – 720 hours	 94.91 97.39 94.91 97.39 0 0
Condenser Volume (credited), ft^3	48,000
ESF Leakage Rate, gpm	11.89
ESF Leakage Iodine Re-Evolution, %	10
ESF Leakage Iodine Release Species, % Elemental Organic	 97 3
SGT System Filter Efficiency, % Aerosol/Particulate Elemental Organic	 98 94 94
Atmospheric Dispersion Factors	Tables 3.1.1 and 3.1.2

Table 3.2.5

Key Parameters Used in Modeling the Control Room for Design Basis Radiological Consequence Analyses

Parameter	Value
Control Room Volume, ft^3	141,900
Normal Ventilation Intake Rate, cfm	3235
CREFS Intake Rate, cfm	900 ± 10% (810 used in analysis)
CREFS Initiation Delay, min	1
CREFS Filter Efficiency, % Elemental Organic Aerosol/Particulate	 89 89 98
Unfiltered Inleakage, cfm	400
Occupancy Factors 0 – 24 hours 24 – 96 hours 96 – 720 hours	 1.0 0.6 0.4
Breathing Rate, m^3/sec	3.5E-04
Atmospheric Dispersion Factors	Table 3.1.1

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

Sincerely,

/ra/

Carl F. Lyon, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures:
1. Amendment No. 234 to DPR-46
2. Safety Evaluation

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