August 29, 2007

Mr. James A. Spina, Vice President
Calvert Cliffs Nuclear Power Plant, Inc.
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 - AMENDMENT RE: IMPLEMENTATION OF ALTERNATIVE RADIOLOGICAL SOURCE TERM (TAC NOS. MC8845 AND MC8846)

Dear Mr. Spina:

The Commission has issued the enclosed Amendment No. 281 to Renewed Facility Operating License No. DPR-53 and Amendment No. 258 to Renewed Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated November 3, 2005, as supplemented by letters dated March 22 and July 17, 2007.

These amendments revise the accident source term in the design basis radiological consequence analyses in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67 which requires licensees who seek to revise their accident source term to apply for a license amendment under 10 CFR 50.90. The proposed accident source term revision replaces the methodology that is based on Technical Information Document (TID)-14844, “Calculation of Distance Factors for Power and Test Reactor Sites,” with the alternate source term methodology described in Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”

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

Sincerely,

/RA/

Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-317 and 50-318

Enclosures:
1. Amendment No. 281 to DPR-53
2. Amendment No. 258 to DPR-69
3. Safety Evaluation

cc w/encls: See next page

August 29, 2007

Mr. James A. Spina, Vice President
Calvert Cliffs Nuclear Power Plant, Inc.
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 - AMENDMENT RE: IMPLEMENTATION OF ALTERNATIVE RADIOLOGICAL SOURCE TERM (TAC NOS. MC8845 AND MC8846)

Dear Mr. Spina:

The Commission has issued the enclosed Amendment No. 281 to Renewed Facility Operating License No. DPR-53 and Amendment No. 258 to Renewed Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated November 3, 2005, as supplemented by letters dated March 22 and July 17, 2007.

These amendments revise the accident source term in the design basis radiological consequence analyses in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67 which requires licensees who seek to revise their accident source term to apply for a license amendment under 10 CFR 50.90. The proposed accident source term revision replaces the methodology that is based on Technical Information Document (TID)-14844, “Calculation of Distance Factors for Power and Test Reactor Sites,” with the alternate source term methodology described in Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”

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

Sincerely,
/RA/
Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-317 and 50-318

Enclosures:
1. Amendment No. 281 to DPR-53
2. Amendment No. 258 to DPR-69
3. Safety Evaluation

cc w/encls: See next page

Package No.: ML072210207
Amendment No.: ML072130521
Tech Spec No.: ML

OFFICE	LPLI-1/PM	LPLI-1/LA	EEEB/BC(A)	SNPB/BC
NAME	DPickett	SLittle	EBrown as signed on	FAkstulewicz as signed on
DATE	8 / 9 / 07	8 /9 / 07	2 / 10 / 06	2 / 21 / 06

OFFICE	AADB/BC	SCVB/BC	OGC	LPL1-1/BC
NAME	MKotzalas as signed on	RDennig	ASilvia	MKowal
DATE	5 / 03 / 07	8 /17/ 07	8 /24 / 07	8 / 29/ 07

OFFICIAL RECORD COPY

DATED: August 29, 2007

AMENDMENT NO. 281 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-53
CALVERT CLIFFS UNIT 1

AMENDMENT NO. 258 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-69
CALVERT CLIFFS UNIT 2

PUBLIC	
LPLI-1 R/F	
M. Kowal	RidsNrrDorlLpl-1
GWilson	RidsNrrDeEeeb
AMendiola	RidsNrrDssSnpb
MKotzalas	RidsNrrDraAadb
ABoatright	
RDennig	RidsNrrDssScvb
HWalker	
S. Little	RidsNrrLASLittle
D. Pickett	RidsNrrPMDPickett
G. Hill (2)	
OGC	RidsOgcMailCenter
ACRS	RidsAcrsAcnwMailCenter
GDentel, R1	

cc: Plant Service list

Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2

cc:

Mr. Michael J. Wallace, President
Constellation Generation Group
750 East Pratt Street
Baltimore, MD 21202

Mr. John M. Heffley
Senior Vice President and
Chief Nuclear Officer
Constellation Generation Group
1997 Annapolis Exchange Parkway
Suite 310
Annapolis, MD 21401

President
Calvert County Board of
Commissioners
175 Main Street
Prince Frederick, MD 20678

Mr. Carey Fleming, Esquire
Sr. Counsel - Nuclear Generation
Constellation Generation Group, LLC
750 East Pratt Street, 17th floor
Baltimore, MD 21202

Mr. Jay S. Gaines
Director, Licensing
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 287
St. Leonard, MD 20685

Mr. R. I. McLean, Manager
Nuclear Programs
Power Plant Research Program
Maryland Department of Natural Resources
580 Taylor Avenue (B wing, 3rd floor)
Tawes State Office Building
Annapolis, MD 21401

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

Ms. Kristen A. Burger, Esquire
Maryland People's Counsel
6 St. Paul Centre
Suite 2102
Baltimore, MD 21202-1631

Ms. Patricia T. Birnie, Esquire
Co-Director
Maryland Safe Energy Coalition
P.O. Box 33111
Baltimore, MD 21218

Mr. Roy Hickok
NRC Technical Training Center
5700 Brainerd Road
Chattanooga, TN 37411-4017

CALVERT CLIFFS NUCLEAR POWER PLANT, INC.

DOCKET NO. 50-317

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 281
Renewed License No. DPR-53

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

   A. The application for amendment by Calvert Cliffs Nuclear Power Plant, Inc. (the licensee) dated November 3, 2005, as supplemented by letters dated March 22 and July 17, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2. of Renewed Facility Operating License No. DPR-53 is hereby amended to read as follows:

2. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 281, are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days following completion of the installation and testing of the plant modifications described in the licensee's letters dated November 3, 2005, March 22 and July 17, 2007.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Mark G. Kowal, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and Technical Specifications

Date of Issuance: August 29, 2007

CALVERT CLIFFS NUCLEAR POWER PLANT, INC.

DOCKET NO. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 258
Renewed License No. DPR-69

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

   A. The application for amendment by Calvert Cliffs Nuclear Power Plant, Inc. (the licensee) dated November 3, 2005, as supplemented by letters dated March 22 and July 17, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2. of Renewed Facility Operating License No. DPR-69 is hereby amended to read as follows:

2. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 258, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days following completion of the installation and testing of the plant modifications described in the licensee's letters dated November 3, 2005, March 22 and July 17, 2007.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Mark G. Kowal, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and Technical Specifications

Date of Issuance: August 29, 2007

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 281 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-53

AMENDMENT NO. 258 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NOS. 50-317 AND 50-318

Replace the following page of the Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
3	3

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

Remove Pages	Insert Pages
iii	iii
iv	iv
v	v
1.1-2	1.1-2
1.1-3	1.1-3
1.1-4	1.1-4
3.4.15-1	3.4.15-1
3.4.15-3	3.4.15-3
3.4.15-4	3.4.15-4
3.7.10-1	- - - - -
3.7.10-2	- - - - -
3.7.11-1	3.7.11-1
3.7.11-2	3.7.11-2
3.9.3-1	3.9.3-1
3.9.3-2	3.9.3-2
5.5-12	5.5-12
5.5-13	5.5-13
5.5-14	5.5-14
5.5-15	5.5-15
5.5-16	5.5-16
5.5-17	5.5-17

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 281 TO RENEWED

FACILITY OPERATING LICENSE NO. DPR-53

AND AMENDMENT NO. 258 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-69

CALVERT CLIFFS NUCLEAR POWER PLANT, INC.

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-317 AND 50-318

1.0 INTRODUCTION

By letter dated November 3, 2005 (Agencywide Documents Access and Management Systems [ADAMS] Accession No. ML053200316), as supplemented by letters dated March 22, 2007 (ADAMS Accession No. ML070870110), and July 17, 2007 (ADAMS Accession No. ML072000313), the Calvert Cliffs Nuclear Power Plant, Inc. (the licensee) submitted a request for changes to the Calvert Cliffs Nuclear Power Plant (CCNPP), Unit Nos. 1 and 2, Technical Specifications (TSs). The letters dated March 22 and July 17, 2007, provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The proposed amendment would implement the alternative source term (AST) methodology for analyzing design-basis accident (DBA) radiological consequences, thereby replacing the existing accident radiological source term that is described in Technical Information Document (TID) TID-14844, “Calculation of Distance Factors for Power and Test Reactor Sites.” As submitted, the license amendment request (LAR) provides the TS changes and DBA radiological consequence analyses associated with the AST, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67 (10 CFR 50.67), “Accident Source Term”, and using the guidance described in Regulatory Guide (RG) 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.” The LAR is for full implementation of the AST with the exception that TID-14844 will continue to be used as the radiation dose basis for equipment qualification and vital area access.

In the 180-day response to Generic Letter (GL) 2003-01, “Control Room Habitability,” the licensee indicated, in its letter dated December 5, 2003 (ADAMS Accession No. ML033440342), that the use of interim compensatory measures, in the form of self contained breathing apparatus (SCBA) and potassium iodide (KI), are assumed in order to mitigate post-DBA dose consequences to plant personnel in the control room. The AST-based re-analyses described in this LAR are intended to remove the reliance on such measures and thereby fulfill the licensee’s commitments stated in their letters dated December 5, 2003, and November 23, 2004 (ADAMS Accession No. ML043380215).

2.0 REGULATORY EVALUATION

The Nuclear Regulatory Commission (NRC) staff evaluated the radiological consequences of the postulated DBAs against the dose criteria specified in 10 CFR 50.67. The applicable criteria are 5 rem Total Effective Dose Equivalent (TEDE) in the control room (CR), 25 rem TEDE at the exclusion area boundary (EAB), and 25 rem TEDE at the outer boundary of the low population zone (LPZ).

The regulatory requirements upon which the NRC staff based its acceptance are Standard Review Plan (SRP) 15.0.1, General Design Criteria (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. Other than the exception discussed in Section 3.5 of this safety evaluation (SE), 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 CCNPP Units 1 and 2 Updated Final Safety Analysis Report (UFSAR) and TSs, as well as consideration for any applicable alternative documentation the licensee may have provided:

- 10 CFR Part 50.67, "Accident source term."
- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants": Criterion 19, "Control room."
- NUREG-0800 SRP Section 2.3.4, "Short-Term Diffusion Estimates for Accidental Atmospheric Releases."
- NUREG-0800 SRP Section 6.3, "Emergency Core Cooling System."
- NUREG-0800 SRP Section 6.4, "Control Room Habitability Systems."
- NUREG-0800 SRP Section 9.4.1, "Control Room Area Ventilation System."
- NUREG 0800 SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Term."
- NUREG-1432, Revision 3, "Standard Technical Specifications - Combustion Engineering Plants."
- RG 1.23, "Onsite Meteorological Programs."
- 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," Rev. 2, March 1987.
- 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.”

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the regulatory and technical analyses performed by the licensee in support of its proposed license amendment, as they relate to the radiological consequences of DBA analyses. 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 analyses. The findings of this SE are based on the descriptions and results of the licensee’s analyses and other supporting information docketed by the licensee.

3.1 Atmospheric Dispersion Estimates

The licensee generated new atmospheric dispersion factors (χ/Q values) for use in evaluating the radiological consequences of six DBAs at CCNPP, located 40 miles south of Annapolis, MD. The licensee used onsite meteorological data for calendar years 1991 through 1998 in the ARCON96 atmospheric dispersion computer model (NUREG/CR-6331 Rev.1, Atmospheric Relative Concentrations in Building Wakes) to calculate CCNPP Units 1 and 2 control room χ/Q values. The licensee used previously generated χ/Q values, as presented in Chapter 2 of the CCNPP UFSAR, for the postulated releases to the EAB and LPZ offsite locations.

3.1.1 Meteorological Data

The licensee used 8 consecutive years of onsite hourly meteorological data collected during calendar years 1991 through 1998 to generate new control room χ/Q values for use in the submitted LAR. These data were provided for NRC staff review in the form of hourly meteorological data files and served as input into the ARCON96 control room atmospheric dispersion computer code. Output from ARCON96 provided control room χ/Q values for six different postulated events. The following are the six events:

- Maximum Hypothetical Accident (MHA) / Loss-of-Coolant Accident (LOCA)
- Fuel-Handling Accident (FHA)
- Main Steam Line Break (MSLB) Accident
- Steam Generator Tube Rupture (SGTR) Accident
- Seized Rotor Event (SRE) / Locked Rotor Accident (LRA)
- Control Element Assembly Ejection Accident (CEAEA) / Control Rod Ejection Accident (CREA)

Wind speed and wind direction data used in the atmospheric dispersion analyses were measured on the CCNPP onsite primary meteorological tower at heights of 10 meters (33 feet) and 60 meters (197 feet) above ground level. Temperature sensors mounted on this same tower provided atmospheric stability data based on temperature difference measurements between the 60-meter and 10-meter interval.

The set of meteorological data used in the submitted LAR atmospheric dispersion analyses (1991 through 1998) was selected from 23 years of data (1982 through 2005) based on the quality assurance provisions pursuant to 10 CFR Part 50 - Appendix B. The combined data recovery of the wind speed, wind direction, and atmospheric stability data was in the upper 90^{th}

percentile during each year of the full data set for measurement levels of 10 meters and 60 meters. Overall, the NRC staff determined there was a data recovery of 98.2%. The licensee noted that the data collection process was based on the guidance provided by RG 1.23, "Onsite Meteorological Programs."

The NRC staff performed confirmatory and quality assurance evaluations of the meteorological data presented. Although staff found numerous instances in which wind directions of 0° were entered into ARCON96 in lieu of 360° for winds blowing from true north, one occurrence in 1995 in which the wind blew from the southeast sector for 102 consecutive hours at the 60 meter height, and a small occurrence of moderate winds under moderately stable conditions (F stability), these events were infrequent (occurred less than 2% of the time) and judged to have an insignificant impact on the use of the data in the current LAR.

Assessment of the wind speed and wind direction data showed similar results from year to year. There was an average wind speed of 2.99 m/s and 4.91 m/s at the 10-meter and 60-meter heights, respectively, for the 8 years of meteorological data presented. Winds predominantly blew from the southwest direction at both measurement levels from year to year. According to the National Oceanic and Atmospheric Administration (NOAA) National Climatic Data Center, these findings are similar to historical data (1950 through 2001) for the Annapolis area with an average ground level wind speed of 2.60 m/s and winds generally blowing from 230° or the southwest direction. Regarding atmospheric stability, the time of occurrence and duration of reported stability conditions were generally consistent with expected meteorological conditions (e.g. neutral and slightly stable conditions predominated during the year with stable and neutral conditions occurring at night and unstable and neutral conditions occurring during the day).

For the reasons noted above, the meteorological data presented for years 1991 through 1998 were found acceptable by the staff and are considered appropriate for atmospheric dispersion estimates for use in the DBA dose assessments performed in support of this LAR.

3.1.2 Control Room Atmospheric Dispersion Factors

The licensee generated new control room χ/Q values for postulated ground level releases from CCNPP Units 1 and 2 for the DBA analyses using guidance provided in RG 1.194. These new atmospheric dispersion estimates were calculated using the ARCON96 onsite atmospheric dispersion computer code. This program, ARCON96, is noted in RG 1.194 as an acceptable methodology for assessing control room χ/Q values for use in DBA radiological analyses. The NRC staff evaluated the applicability of the ARCON96 model and determined that there are no unusual siting, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude use of this model in support of this LAR for the CCNPP.

The wind speed, wind direction, and atmospheric stability measured at the 10-meter and 60-meter heights served as input for these calculations. Releases of radioactivity from the six DBAs were assumed to discharge to the environment via six different sets of sources (12 sources in total): Units 1 and 2 containment, Units 1 and 2 Atmospheric Dump Valve (ADV), Units 1 and 2 Containment Outage Doors (COD), Units 1 and 2 Main Steam Gooseneck (MSG), Units 1 and 2 Ventilation Stack (VS), and Units 1 and 2 Refueling Water Tank (RWT). The release heights for each set of these sources are 29.7 meters, 17.2 meters, 2.5 meters, 17.1 meters, 48.3 meters, and 14.7 meters, respectively. Essentially, all releases were assumed to

occur at ground level for the purpose of atmospheric dispersion analyses. The primary onsite receptors, used for control room atmospheric dispersion evaluations, were the Auxiliary Building West Road (WR) inlets, the Turbine Building (TB), and the Access Controls (AC11 and AC13) on the roof of the Auxiliary Building. The licensee used the taut string methodology described in RG 1.194 for containment releases. Initial sigma-y (6.99) and sigma-z (5.18) values were entered only for containment releases which were modeled as a diffuse source. All other releases were modeled as point sources.

The NRC staff qualitatively reviewed the inputs to the ARCON96 calculations and found them generally consistent with site configuration drawings and staff practice. Additionally, the staff performed some confirmatory analyses of the data and found them acceptable. Thus, the NRC staff has concluded that the χ/Q values for CCNPP releases to the control room as presented in Table 3.1.2, which is attached to this SE, are acceptable for use in the DBA control room dose assessment performed in support of this LAR.

3.1.3 EAB and LPZ Atmospheric Dispersion Factors

The licensee used previously generated EAB and LPZ χ/Q values at downwind distances of 1150 meters (0.71 miles) and 3219 meters (2 miles), respectively, for the six postulated DBAs. The licensee extracted the EAB value from Chapter 2.3.6 and the LPZ values from Chapter 14.24.3 of the CCNPP UFSAR. For the current LAR, the licensee increased the 0-2 hour EAB and LPZ χ/Q values of 1.30E-4 sec/m^3 and 3.30E-5 sec/m^3, respectively, from the value used in the CCNPP UFSAR, Chapter 2.3.6, "Calculation of Incident and Routine Long-Term Relative Concentrations," due to an adjustment for a ventilation stack release rather than a containment release. Note that the resulting values of 1.44E-4 sec/m^3, at the EAB, and 3.39E-5 sec/m^3, at the outer boundary of the LPZ are slightly more conservative than the currently approved design basis values.

3.2 Radiological Consequences of DBAs

To support the proposed implementation of an AST, the licensee analyzed the radiological dose consequences and provided all major inputs and assumptions for the six postulated DBAs. In order to revise the CCNPP licensing basis to incorporate a full implementation of the AST, RG 1.183 Position 1.2.1 specifies that the DBA LOCA must be re-analyzed using the appropriate guidance therein. In accordance with RG 1.183 guidance, the licensee re-analyzed the six postulated DBAs, which includes the design-basis LOCA at CCNPP. In addition to these six DBA analyses, the licensee also analyzed the radiological consequences for the following two incidents:

- Waste Gas Incident (WGI)
- Waste Processing Incident (WPI)

These two incidents, however, are not required for approval of an LAR to implement AST-based analyses, nor do they determine licensing basis parameters for CCNPP. Therefore, by letter dated March 22, 2007, the licensee withdrew the WGI and WPI from the list of incidents considered in their LAR. Thus, the WGI and WPI were not reviewed for this SE.

The licensee's application provides the results of the radiological consequence analyses for the postulated DBAs to show compliance with 10 CFR 50.67 dose acceptance criteria, or fractions

thereof, as defined in SRP 15.0.1, for doses offsite and in the control room. For full implementation of the AST DBA analysis methodology, 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. The subject LAR is considered a full implementation of the AST.

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 Emergency Core Cooling System (ECCS) 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 accident analyses postulating fuel damage, and in accordance with RG 1.183 guidance, the licensee calculated the core isotopic inventory available for release using the SAS2H/ORIGEN-S isotope generation and depletion computer code, and then multiplied the isotopic specific activities by the relevant power level and release fractions. The 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.

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 review of the DBA dose assessment performed by the licensee to support the LAR submittal of November 3, 2005, including all supporting supplements.

3.2.1 Maximum Hypothetical Accident/LOCA

The current CCNPP design-basis LOCA analysis is based on the traditional accident source term described in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." The current licensing basis radiological consequence analysis for the postulated LOCA is provided in the CCNPP UFSAR Chapter 14.24, "Maximum Hypothetical Accident." To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and control room radiological consequences of the postulated LOCA. This analysis was performed to demonstrate that the engineered safety features (ESFs) designed to mitigate

the radiological consequences at CCNPP will remain adequate after implementation of the AST.

Included in the LOCA analysis are the assumptions, parameters, and newly calculated offsite and control room doses associated with implementing the AST methodology. The licensee cites 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 is detailed in Appendix A of that document.

3.2.1.1 Activity Source

For the LOCA analysis, the licensee assumed that the core isotopic inventory, that is available for release into the containment atmosphere, is based on maximum full power operation of the core at 2,754 MWth, or 1.02 times the current licensed thermal power level of 2,700 MWth, in order to account for the ECCS evaluation uncertainty. Additionally, current licensed values for fuel enrichment and burnup are assumed when determining the core isotopic inventory.

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 1.1.83, Tables 2 ands 4, respectively. Also, consistent with RG 1.183 guidance, the licensee assumed that the radioactive iodine speciation released from failed fuel is 95.00% aerosol (particulate), 4.85% elemental, and 0.15% organic. Whereas, the radioactive iodine speciation released from the steam generators (SGs) is 97.00% elemental and 3.00% organic.

3.2.1.2 Transport Methodology and Assumptions

For releases into containment, the licensee credits the CCNPP fan coolers for mixing air and entrained activity between the sprayed and unsprayed regions of containment. CCNPP has three 110,000 cubic feet per minute (cfm) cooling units in normal operation, with a fourth started at low speed (55,000 cfm) upon receiving a Safety Injection Actuation Signal (SIAS). However, following the loss of offsite power (LOOP) and assumed worst-case single failure of an emergency diesel generator (EDG), the licensee conservatively assumes only two cooling units, at the 55,000 cfm low speed, are available for mixing beginning 60 seconds after the start of the accident.

The NRC staff has reviewed the licensee's assessment of the following potential post-LOCA activity release pathways:

- Containment Surface Leakage
- Ventilation Stack Release
- RWT Release of ECCS Leakage
- Hydrogen Purge Line Leakage

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

- Containment Shine
- Plume Shine
- Control Room Filter Shine

The following sections detail the NRC staff review of the licensee's analysis of these post-accident contributors to both control room and offsite dose.

3.2.1.2.1 Containment Surface Leakage

The current CCNPP design basis containment leak rate, L_a, is equal to 0.2 percent of containment air weight per day (% per day), at containment peak pressure, as expressed in CCNPP TS 5.5.16. The licensee proposes to reduce this L_a to a value of 0.16 % per day for the first 24 hours of the accident, with a subsequent reduction to half of this value for the remainder of the accident duration, which is consistent with the guidance of RG 1.183.

The airborne activity in the CCNPP containment following the postulated LOCA is mitigated by natural deposition of fission products in aerosol form, containment filtration, and removal by the containment spray system. The following subsections discuss the evaluation of the credit that is taken for activity mitigation by these processes.

The use of models for the various mechanisms for iodine removal, when more than one is used simultaneously for the same iodine species in a dose analysis, should consider the effect of one model on the others. Because each model used by the licensee to simulate the removal of activity does not necessarily account for removal through the other models, the use of the referenced natural deposition, containment filtration, and spray removal models in the same region of containment, during the same time period, is recognized as potentially non-conservative. Although natural deposition, the containment iodine removal system (IRS), and containment sprays are all acting on the overall in-containment aerosol and elemental iodine source term, the total effect from each of these removal mechanisms is not the same as would be found by simply adding the removal coefficients for each model for a given time period. Therefore, this treatment of containment removal modeling is generally found to be unacceptable to the NRC staff. However, with regard to natural deposition, it is understood that in the presence of the containment IRS and containment sprays the additional effect of crediting natural deposition is minimal. It can also be concluded, from the staff's confirmatory calculations based on the licensee's submitted dose analyses, that adequate conservatism was used in modeling the containment IRS and containment spray systems such that the non-conservative effect of the two systems on one another is sufficiently accounted for. In addition, as discussed in the following sections, there is adequate justification that both the containment IRS and containment spray systems will indeed be available to mitigate post-LOCA activity leakage. Therefore, for this amendment request, the staff does not deem it necessary for the licensee to recalculate the iodine removal, and subsequent resulting LOCA dose consequences, using a more conservative modeling of iodine removal mechanisms. The staff's confirmatory analysis indicates that, while the inclusion of the effects of natural deposition in conjunction with ESF iodine removal mechanisms is, in theory, non-conservative, the overall iodine removal, as determined by the licensee in this case, is conservative and is therefore acceptable.

3.2.1.2.1.1 Containment Iodine Removal System (IRS)

The IRS at CCNPP has three filter units, called iodine removal units (IRUs), each with a capacity to handle 50% of the required air flow. Each IRU consists of activated charcoal filters preceded by HEPA filters and has a 20,000 cfm ±10% flowrate. The CCNPP IRS is designed such that one IRU is connected to one of two buses, while the other two IRUs are connected to the other bus. As a result, the licensee assumes that only one IRU is automatically initiated at 63 seconds, accounting for delays associated with a LOOP followed by a worst-case single failure of an EDG. Then the licensee assumes that a second IRU is manually initiated at 20 minutes. Therefore, at 63 seconds the filtration efficiencies credited are 45% for aerosol and elemental, and 15% for organic iodine forms, and after 20 minutes these efficiencies are increased to 90% for aerosol and elemental, and 30% for organic. The licensee's treatment of the IRS IRU initiation is conservative and consistent with current CCNPP design basis assumptions, as shown in CCNPP UFSAR 14.24.3, and is, therefore, acceptable to the NRC staff.

3.2.1.2.1.2 Natural Deposition

The licensee's analysis assumed removal of airborne activity in aerosol form by natural deposition in containment following the postulated LOCA using Powers' simplified natural deposition model in the dose consequences computer code described in NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," and its supplements. Powers' simplified natural deposition model is described in NUREG/CR-6189, "A Simplified Model of Aerosol Removal By Natural Processes in Reactor Containments." The licensee conservatively used the 10th percentile confidence interval (90 percent probability) removal values implemented in the RADTRAD code. The Powers natural deposition model was derived by correlation of results of Monte Carlo uncertainty analyses of detailed models of aerosol behavior in the containment under accident conditions. The NRC staff finds that the use of this model, as implemented in the NRC computer code, RADTRAD, is acceptable, as discussed in RG 1.183.

3.2.1.2.1.3 Containment Sprays

The containment spray system at CCNPP is an ESF system. When used in conjunction with fan coolers, it is designed to suppress containment pressure and remove fission products in the containment atmosphere following a postulated LOCA. The licensee credits the fan coolers for mixing air between the sprayed and unsprayed regions of containment. As discussed in Section 3.2.1.2, two 55,000 cfm cooling units, assumed to start at 60 seconds, are credited in the licensee's LOCA analysis, which accounts for a single EDG failure. At the combined flowrate, these fan coolers will provide two turnovers of the entire containment air volume in less than 1 hour. Therefore, the licensee has shown that adequate mixing between sprayed and unsprayed regions of the containment atmosphere is provided in accordance with the guidance provided in RG 1.183, which assumes two turnovers of the unsprayed region per hour.

The CCNPP containment spray system, which is redundant with the containment air cooling system, consists of two 50% capacity pumps, two spray headers and nozzles, two heat exchangers, and all associated valves, piping, and instrumentation. The pumps discharge borated water from the RWT through the heat exchangers to the spray headers and nozzles located in the containment. The spray nozzles are arranged in the headers to give complete

spray coverage at the containment horizontal cross-sectional area. The sprayed volume of the containment is 72.73% of the total free volume of the containment. After a postulated LOCA, suction for the containment spray and safety injection systems will be taken from the RWT. The water introduced into the containment in this manner will mix with water from the primary reactor coolant system (RCS). The resultant mixture will be used in the spray and injection systems only after the inventory of the RWT is nearly depleted.

Each pump of the CCNPP containment spray system is independently capable of delivering 1,400 gallons per minute (gpm) (187.15 cfm) of borated water from the RWT during the injection phase and 1,350 gpm (180.47 cfm) of mixed coolant from the containment sump during the recirculation phase into 72.73% of the containment free volume. The licensee assumed a flow of 180 cfm (1,346.49 gpm) throughout the containment spray system operation for determining the fission product removal coefficients by spray. The licensee conservatively credits only one out of two spray pumps for removal of iodine activity in containment.

3.2.1.2.2 Ventilation Stack Release

Consistent with post-LOCA containment surface leakage, the airborne activity in the CCNPP containment, released through the vent stack following a postulated LOCA, is also mitigated by natural deposition of fission products in aerosol form, containment filtration, and removal by the containment spray system. Additionally, for releases from containment through the CCNPP Ventilation Stack pathway, the Containment Penetration Room Emergency Ventilation System (PREVS) is credited to further reduce the activity concentration released to the environment. The following subsections discuss the evaluation of the credit that is taken for activity mitigation by these processes.

3.2.1.2.2.1 Containment IRS

The activity transport description and technical evaluation discussed for containment surface leakage in Section 3.2.1.2.1.1 above is also applicable to this activity release pathway.

3.2.1.2.2.2 Natural Deposition

The activity transport description and technical evaluation discussed for containment surface leakage in Section 3.2.1.2.1.2 above is also applicable to this activity release pathway.

3.2.1.2.2.3 Containment Sprays

The activity transport description and technical evaluation discussed for containment surface leakage in Section 3.2.1.2.1.3 above is also applicable to this activity release pathway.

3.2.1.2.2.4 Containment PREVS

Commensurate with guidance in RG 1.194, which allows for the total containment release to be apportioned between exposed and enclosed building surfaces, the licensee assumes a conservative bypass fraction from the containment to the penetration rooms of 28%. Thus, 0.0448% of the containment air, by weight, leaks from the containment to the auxiliary building penetration rooms per day for the first 24 hours, and 0.0224% per day thereafter. Based on the ratio of containment surface enclosed by the auxiliary building to the total containment surface

area, the licensee states that at least 28.7% of the containment leakage would be into the penetration rooms. Therefore, based on this calculation and other conservatisms, the assumption that 28% of the containment leakage would be into the penetration rooms is conservative and acceptable to the staff.

Following a LOCA at CCNPP, the licensee states that a containment isolation signal will start both of two blower units associated with the PREVS. Each of these units has a design flowrate of 2000 cfm ± 10%. For each of these units the filtration efficiencies credited are 90% for aerosol and elemental, and 30% for organic iodine forms, per CCNPP TS 5.5.11; therefore credit for this system in mitigating containment releases through the ventilation stack is consistent with the CCNPP licensing basis.

3.2.1.2.3 RWT Release of ECCS Leakage

ESF systems that recirculate sump water outside of containment are assumed to leak during their intended operation. At CCNPP, the licensee has identified such potential unmonitored release pathways. These pathways are the result of post-LOCA isolation valve leakage in the Safety Injection or Containment Spray system recirculation lines to the RWT, which is vented directly to the environment. Specifically, the licensee states that there are two (2) pathways from which this coolant leakage can take place; (1) two valves in series on the minimum flow recirculation line header, and (2) the valve from the containment spray pumps. Combined ECCS leakage into the RWT from these two pathways is limited, by the CCNPP TS, to 1000 cm^3/hour. This rate is, in turn, doubled in accordance with RG 1.183 guidance, to account for valve degradation between testing.

The licensee assumes that, with the exception of iodine, all radioactive material in the recirculating coolant is retained in the liquid phase, and, in accordance with RG 1.183, the chemical form of the released iodine activity is assumed to be 97.00% elemental and 3.00% organic. The licensee uses a constant enthalpy assumption and an associated equation to calculate the flashing fraction of leaked coolant. The licensee determined this value to be less than 3%; however, in accordance with RG 1.183 guidance, a conservative flashing fraction of 10% is used. The licensee assumes that the 10% flashed activity in the RWT is vented to the environment at a rate of 4.2 cfm.

3.2.1.2.4 Hydrogen Purge Line Leakage

The CCNPP containment is routinely purged, so it was necessary for the licensee to consider the dose contribution of releases from this pathway. It is assumed that the purge system is isolated within 30 seconds, which is prior to the gap or early in-vessel release phases of the LOCA. Therefore, only releases of the TS controlled equilibrium activity in the primary RCS have to be considered. The equilibrium activity assumed is equal to 0.5 μCi/gm dose equivalent (DE) I-131, which represents the proposed reduction in the current licensing basis activity concentration of 1.0 μCi/gm Dose Equivalent (DE) I-131. Due to this being a flashed coolant release, the licensee assumes that the speciation of the iodine activity available for release is 97.00% elemental and 3.00% organic, indicating that the aerosol iodine remains entrained in the coolant.

The licensee assumes it takes a maximum of 30 seconds to isolate the purge valve. This time is based on 2.4 seconds for containment pressure buildup, instrument response, and SIAS

delay, 10 seconds for EDG startup, and 15 seconds for valve stroke time. The licensee includes an additional margin of 2.6 seconds is included for conservatism. The hydrogen purge line is assumed to be released through the vent stack; therefore, PREVS is credited to further reduce the activity concentration released to the environment. For this release the PREVS filtration efficiencies credited are 90% for aerosol and elemental, and 30% for organic iodine forms, per CCNPP TS 5.5.11; therefore credit for this system in mitigating RCS activity releases through the hydrogen purge line is consistent with the CCNPP licensing basis.

3.2.1.3 Direct Shine Dose Methodology

3.2.1.3.1 Containment Shine

For the calculation of internal containment cloud shine dose to the control room, the licensee assumes that the source term and associated release fractions described in RG 1.183 are instantaneously released into the containment volume at the beginning of the accident. For calculation of the internal containment cloud shine dose contribution to the 30-day control room dose, this conservative simplification is acceptable to the NRC staff.

The licensee used the MicroShield code to calculate the time dependent dose rates in the control room that result from shine from the containment activity cloud. Using a selection of 17 time steps, the licensee calculated individual dose rates, and then integrated the results to determine the total shine dose contribution from the containment cloud. Although decay and the subsequent formation of daughter products were accounted for, the licensee conservatively took no credit for removal of activity from the containment cloud by leakage or other removal mechanisms.

3.2.1.3.2 Plume Shine

For the calculation of shine dose to the control room resulting from an activity plume external to containment, the licensee assumes that the source term and associated release fractions described in RG 1.183 are instantaneously released into the containment volume at the beginning of the accident. The volume of the plume is then characterized by an instantaneous release from containment at the newly proposed design basis containment leak rate, L_a, of 0.16% per day for the first day, and then reduced to half that value, or 0.08% per day, for the remaining 29 days of the 30-day accident duration. This is an acceptable assumed reduction that is based on a time dependent containment pressure decrease. For calculation of the shine dose contribution to the 30-day control room dose resulting from the plume external to containment, this conservative simplification is acceptable to the staff.

The licensee used the MicroShield code to calculate the time dependent dose rates in the control room that result from the activity plume external to containment. Using a selection of 11 time steps, the licensee calculated individual dose rates, and then integrated the results to determine the total shine dose contribution from the external plume. Although decay and the subsequent formation of daughter products were accounted for, the licensee assigned all leakage to the external plume and conservatively took no credit for other activity removal mechanisms.

The licensee also calculated an activity dilution factor to model a plume incident, horizontally, on the control room. The dilution factor was calculated by dividing the assumed plume length that

will travel past the control room dose receiver, 35 meters, by the distance that the plume is expected to travel during the 30-day accident duration, based on an assumed wind velocity. Per design analysis CA06012, “CRHVAC Atmospheric Dispersion Coefficient Calculations,” a wind velocity of 2.99 meters per second, averaged over 8 years, was determined; however, for calculating the dilution factor the licensee assumed a velocity of 0.3 meters per second. This treatment of the plume dilution is a simplification due to uncertainties in wind velocity, plume geometric and dynamic characteristics, and activity concentration. While a more robust modeling of atmospheric dispersion could have been used, the staff finds the licensee’s approach acceptable because the plume shine is a characteristically small contributor to total dose and the plume activity concentration was conservatively derived.

3.2.1.3.3 Control Room Filter Shine

For the calculation of shine dose to the control room resulting from the accumulation of activity on control room filters, the licensee assumes that the source term and associated release fractions described in RG 1.183 are instantaneously released into the containment volume at the beginning of the accident. The iodine speciation specified in RG 1.183 was assumed, and containment sprays and the containment IRS are both accounted for in assessing what activity is released and available to deposit on the control room filter. For deposition of activity on the control room filter, the licensee assumes that all nuclides other than noble gases are in particulate form. Noble gases are not assumed to be removed by the control room filters, and are, therefore, ignored for this dose contributor.

The licensee used the MicroShield code to calculate the time dependent dose rates in the control room that result from the activity loaded control room filter. A geometry was described to model the control room filter location with relation to the nearest hypothetical control room dose receiver, 1 inch away from the common 2-foot thick concrete control room wall, and on the centerline of the filter.

The licensee’s treatment of this shine dose contributor is conservative with regard to the activity release and geometry used for MicroShield modeling. The MicroShield point-kernel method used here is appropriate for this case, because the geometry avoids oblique angles and does not require excessive use of buildup factors to determine dose. Therefore, the NRC staff finds the licensee’s model of the control room filter shine dose to be conservative and acceptable.

3.2.1.4 Conclusion

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 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 2.0 of this SE and with those stated in the CCNPP UFSAR as design bases. 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 confirmed the licensee’s dose results. The major parameters and assumptions used by the licensee and found acceptable to the staff are presented in Table

3.2.1 (attached). The results of the licensee's design basis radiological consequence calculation are provided in Table 3.2 (attached). The EAB, LPZ, and CR doses estimated by the licensee for the LOCA were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.2.2 Fuel-Handling Accident (FHA)

The current CCNPP design basis FHA analysis is based on the traditional accident source term described in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." The CCNPP licensing basis analysis is presented in UFSAR Chapter 14.18, "Fuel Handling Incident." To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and control room radiological consequences of the postulated FHA. This reanalysis was performed to demonstrate that the engineered safety features (ESFs) designed to mitigate the radiological consequences at CCNPP will remain adequate after implementation of the AST.

The licensee submitted the AST-based analysis of the FHA as part of the LAR. Included in this analysis are the assumptions, parameters, and newly calculated offsite and control room doses associated with implementing the AST methodology. The licensee cites RG 1.183 as providing the primary radiological analysis assumptions for their analysis of the postulated design-basis FHA. Specifically, the NRC staff's guidance is detailed in Appendix B of that document.

3.2.2.1 Activity Source

For the FHA analysis, the licensee assumed that the core isotopic inventory is based on maximum full power operation of the core at 2,754 MWth, or 1.02 times the current licensed thermal power level of 2,700 MWth, in order to account for the ECCS evaluation uncertainty. An additional pin power peaking factor of 1.70 is applied. The licensee states that the current CCNPP Core Operating Limits Report (COLR) identifies total integrated radial peaking factors of less than, or equal to, 1.65 for both Unit 1 and 2. Therefore, the assumed pin power peaking factor of 1.70 is conservative. Additionally, the licensee accounts for current licensed values for fuel enrichment and burnup when determining the core isotopic inventory. The fraction of core isotopic activity assumed to be available for release from the gap of failed fuel (i.e., fuel experiencing cladding failure as a result of the drop) is provided in Table 3 of RG 1.183. To account for gap fraction uncertainty in fuel that does not meet the criteria specified in footnote 11 of RG 1.183, the licensee multiplied these gap fractions by a factor of two. This is a conservative approach that is acceptable to the NRC staff.

Per CCNPP Technical Requirements Manual (TRM) 15.9.1, fuel movement can occur no earlier than 100 hours following shutdown. Therefore, administratively, any fuel that could potentially be involved in the postulated FHA will have experienced a minimum of 100 hours of decay time. However, for conservatism, the licensee assumed only 72 hours of decay time for the accident analysis. The licensee assumed that 176 pins will be damaged as the result of the postulated FHA, and thus release all of their available gap activity over a 2-hour period. This is consistent with the current CCNPP licensing basis, as described in CCNPP UFSAR Section 14.18, and the guidance expressed in RG 1.183.

3.2.2.2 Transport Methodology and Assumptions

As analyzed for CCNPP, the most limiting FHA is a drop of an assembly in the SFP during inspection and reconstitution. When assemblies are placed on rack spacers with their upper end fitting removed, the licensee postulates that there will be 20.4 feet of water coverage between the top of the pin and surface of the water. The licensee calculated that iodine activity released subsequent to the postulated drop will be removed by an overall aerosol and elemental iodine decontamination factor (DF) of 120, which is associated with the minimum water coverage that was determined. This DF value was interpolated based on an overall iodine DF of 200 being associated with water coverage of 23 feet, as described in RG 1.183, Appendix B. The licensee also determined that the difference in DF for elemental and organic iodine species results in the activity released from the SFP being composed of approximately 82% elemental and 18% organic iodine. Noble gas activity is assumed to be released from the SFP water without experiencing any reduction. The DF calculation by the licensee is conservative and acceptable to the staff.

Releases of activity from the SFP following the FHA are assumed to be released unfiltered through the plant vent at a rate of 100 cfm for 30 days. This is a nominal value used in the RADTRAD code, when coupled with an assumed 1 ft^3 containment volume to model an immediate release of the activity from the damaged fuel. This treatment ensures no credit is taken for holdup or dilution in the containment volume. This path was determined by the licensee to be the most limiting release point for the design basis drop in the SFP.

3.2.2.3 Conclusion

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 FHA are a TEDE of 6.3 rem at the EAB for any 2 hours, 6.3 rem at the outer boundary of the LPZ and 5 rem in the control room for the duration of the accident. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the CCNPP UFSAR as design bases. The staff also performed independent calculations to verify the conservatism of certain parameters used by the licensee. The major parameters and assumptions used by the licensee and found acceptable to the staff are presented in Table 3.2.2 (attached). The results of the licensee's design basis radiological consequence calculation are provided in Table 3.2 (attached). The EAB, LPZ, and CR doses estimated by the licensee for the FHA accident were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.2.3 Main Steam Line Break (MSLB) Accident

The current CCNPP design basis MSLB analysis is based on the traditional accident source term described in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." The CCNPP licensing basis analysis is presented in UFSAR Chapter 14.14, "Steam Line Break Event." To support implementation of the AST, the licensee reanalyzed the offsite and control room radiological consequences of the postulated MSLB. This analysis was performed to demonstrate that the ESF designed to mitigate the radiological consequences at CCNPP will remain adequate after implementation of the AST.

The licensee submitted the AST-based analysis of the MSLB accident as part of the LAR. Included in this analysis are the assumptions, parameters, and newly calculated offsite and control room doses associated with implementing the AST methodology. The licensee cites RG 1.183 as providing the primary radiological analysis assumptions for their analysis of the postulated design-basis MSLB accident. Specifically, the staff's guidance is detailed in Appendix E of that document.

The licensee has defined the design-basis MSLB accident as the pre-trip, guillotine-type rupture of a main steam line outside containment. Specifically for CCNPP, this is assumed to occur in the Main Steam Piping Room (MSPR), between a SG and main steam isolation valve (MSIV). The radiological consequences of an MSLB outside containment will bound the consequences of a break inside containment. Thus, only an MSLB outside of containment is considered with regard to the radiological consequences. Fission products are introduced into the nuclear steam supply system (secondary side) through steam generator tube leakage, also referred to as primary-to-secondary leakage.

3.2.3.1 Activity Source

With regard to a postulated release following a design-basis MSLB, the licensee considered the following three cases in order to determine the maximum offsite and control room dose:

- Failed Fuel Case: As shown in CCNPP UFSAR Chapter 14.14, the current MSLB accident analysis assumes 1.35% fuel failure; however, the revised analysis for this LAR assumes a reduced value of 0.80%. The licensee bases this change on analyses CA06383, "Calvert Cliffs Units 1 and 2 Pre-Trip Steam Line Break Event," and CA06382, "Calvert Cliffs Unit 1 Cycle 17 Post-Trip Steam Line Break Event," which calculate no fuel damage resulting from the postulated MSLB. For this MSLB accident case, the initial thermal power is assumed to be 2,754 MWth, which is a factor of 1.02 times the current licensed thermal power of 2,700 MWth, in order to account for the ECCS evaluation uncertainty. An additional pin power peaking factor of 1.70 is applied. The licensee states that the current CCNPP COLR identifies total integrated radial peaking factors of less than, or equal to, 1.65 for both Unit 1 and 2. So, the assumed pin power peaking factor of 1.70 is conservative. Additionally, the licensee accounts for current licensed values for fuel enrichment and burnup when determining the core isotopic inventory. The fraction of isotopic activity assumed to be available for release from the gap of fuel that experiences cladding failure is that provided in Table 3 of RG 1.183, and then multiplied by a factor of two. This is a conservative approach that is found acceptable to the staff to account for gap fraction uncertainty in fuel not meeting the criteria specified in footnote 11 of RG 1.183.

- Preaccident Iodine Spike (PIS) Case: For this case, the licensee assumes that a reactor transient has occurred prior to the postulated MSLB and has raised the primary RCS iodine concentration to 60 times the TS 3.4.15 limit, as is consistent with the guidance of RG 1.183, when no fuel failure is assumed. The primary RCS iodine concentration that is assumed, as part of the changes requested by this LAR, is 0.5 μCi/gm DE I-131. This identifies a requested reduction in the current licensing basis primary RCS activity concentration of 1.0 μCi/gm DE I-131.

- Concurrent Iodine Spike (CIS) Case: For this case, the licensee assumes that the RCS transient associated with the MSLB causes an iodine spike in the primary RCS. It is assumed that the iodine release rate from the fuel rods to the primary RCS increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value, which is consistent with the guidance of RG 1.183, when no fuel failure is assumed. The licensee conservatively assumed a spike duration equal to the 9-hour shutdown cooling (SDC) time, then increased the activity by a factor of 9/8, to account for this deviation from the RG 1.183 guidance, which states that the duration should be 8 hours.

In addition, for the three cases considered, the licensee also assumes that the TS maximum secondary coolant iodine concentration is available for release, as inferred by RG 1.183 guidance. As shown in TS 3.7.14, the maximum secondary coolant iodine concentration at CCNPP is 0.1 μCi/gm DE I-131. Consistent with RG 1.183 guidance, the licensee assumes that the radioactive iodine speciation released from failed fuel is 95.00% aerosol (particulate), 4.85% elemental, and 0.15% organic. Whereas, the radioactive iodine speciation released from the SGs is 97.00% elemental and 3.00% organic.

After analyzing these three cases, the licensee determined that the dose consequences from the PIS case are bounded by the case that assumes fuel failure. However, because the dose acceptance criteria as shown in RG 1.183 are lower for the CIS case, the dose consequences from the CIS case are also relevant to this DBA analysis.

3.2.3.2 Transport Methodology and Assumptions

The current licensing basis at CCNPP restricts primary-to-secondary leakage rate to a TS limited rate of 200 gallons per day (gpd) for both SGs. For the MSLB accident analyses, the licensee assumes RCS leakage at this rate until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C, as specified by RG 1.183. In the analysis NEU-98-027, ES199800165-000, "Engineering Evaluation to Determine the Time Required to Cool the RCS to 212°F During a Main Steam Line Break Scenario," the licensee calculates this to occur within 9 hours, with the assumption of a single failure concurrent with a LOOP. A partitioning factor of 1, or unity, is used for all discharged activity. Therefore, no credit is taken for steam partitioning in the licensee's MSLB accident analysis.

For this LAR, the licensee is seeking to reduce the maximum TS primary RCS activity from 1.0 μCi/gm DE I-131 to 0.5 μCi/gm DE I-131. The licensee assumes that all primary-to-secondary activity leakage is released through the affected SG, directly to the environment, assuming 100% flashing, and without credit for holdup in the SG or the MSPR. This treatment takes no credit for possible leakage through the unaffected SG that would subsequently have an opportunity to partition. The licensee's method maximized the dose consequence in accordance with RG 1.183 guidance, by apportioning the leakage between affected and unaffected SGs in the most conservative manner, and is, therefore, acceptable to the NRC staff.

The licensee accounts for the release of activity in the secondary coolant by immediately releasing the maximum secondary coolant activity allowed by the TS, 0.1 μCi/gm DE I-131, directly to the environment. To accomplish this "immediate" release, the licensee assumed that the total coolant mass in both SGs is released at a rate of 4000 cfm. In the MSLB analysis

submitted by the licensee, Calc CA06452 (ML0532003030), it is stated that the SG volume is 4420.04 ft^3. Assuming this coolant volume and a 4000 cfm release rate, all of the available activity will be released in approximately 1.1 minutes. If the accumulated dose, during the early time period prior to 1 minute, was calculated using a greater number of time steps, as allowed in the RADTRAD code, a higher dose would be assessed. However, because the resulting dose increase due to this timing refinement will be negligible in this case, and because of the relatively small dose contribution from this pathway, the NRC staff finds this to be a sufficiently conservative and acceptable treatment.

3.2.3.3 Conclusion

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 MSLB with fuel failure are a TEDE of 25 rem at the EAB for any two hours, 25 rem at the outer boundary of the LPZ and 5 rem for access and occupancy of the control room for the duration of the accident. For the MSLB assuming a CIS, the accident-specific dose acceptance criteria are a TEDE of 2.5 rem at the EAB for any 2 hours, 2.5 rem at the outer boundary of the LPZ and 5 rem for the control room. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the CCNPP UFSAR as design bases. The staff also performed an independent calculation of the dose consequences of the MSLB accident using the licensee's assumptions for input to the RADTRAD computer code. The staff's calculation confirmed the licensee's dose results. The major parameters and assumptions used by the licensee and found acceptable to the staff are presented in Table 3.2.3 (attached). The results of the licensee's design basis radiological consequence calculation are provided in Table 3.2 (attached). The EAB, LPZ, and CR doses estimated by the licensee for the MSLB accident were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.2.4 Steam Generator Tube Rupture (SGTR) Accident

The current CCNPP design basis SGTR accident analysis is based on the traditional accident source term described in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." The CCNPP licensing basis analysis is presented in UFSAR Chapter 14.15, "Steam Generator Tube Rupture Event." To support implementation of the AST, the licensee reanalyzed the offsite and control room radiological consequences of the postulated SGTR. This analysis was performed to demonstrate that the ESFs designed to mitigate the radiological consequences at CCNPP will remain adequate after implementation of the AST.

The licensee submitted the AST-based analysis of the SGTR accident as part of the LAR. Included in this analysis are the assumptions, parameters, and newly calculated offsite and control room doses associated with implementing the AST methodology. The licensee cites RG 1.183 as providing the primary radiological analysis assumptions for their analysis of the postulated design-basis SGTR accident. Specifically, the NRC staff's guidance is detailed in Appendix F of that document.

At CCNPP, the licensee has analyzed the limiting SGTR accident as a complete double-ended tube break that is postulated to occur due to a complete failure of a tube-to-sheet weld or the

rapid propagation of a circumferential crack. The SGTR allows primary coolant to leak into the secondary system via the SG. The primary coolant transfer causes the pressurizer level to decrease, provided that the tube leak rate exceeds the charging pump capacities and causes the level in the affected SG to increase. In the case of this double ended tube rupture, the leak rate far exceeds the charging pump capacities and, consequently, the pressurizer level will decrease. The decrease in the pressurizer level and the inability of the heaters to maintain pressurizer pressure causes the RCS pressure to decrease. The drop in pressure will cause a reactor trip on Thermal Margin/Low Pressure (TM/LP), ensuring that the Departure from Nucleate Boiling (DNB) Specified Acceptable Fuel Design Limit (SAFDL) is not exceeded. Therefore, the licensee assumes no fuel failure resulting from a design-basis SGTR accident at CCNPP.

3.2.4.1 Activity Source

The licensee determined that there is no fuel failure associated with the design-basis SGTR accident. As a result, the licensee considered the following two cases of postulated activity release following a design basis SGTR, in order to determine the maximum offsite and control room dose:

- PIS Case: For this case, the licensee assumes that a reactor transient has occurred prior to the postulated SGTR and has raised the primary RCS iodine concentration to 60 times the TS 3.4.15 limit, which is consistent with the guidance of RG 1.183 when no fuel failure is assumed. The TS limited primary RCS iodine concentration that is assumed, as one of the changes requested by this LAR, is 0.5 μCi/gm DE I-131. This identifies a requested reduction in the current licensing basis primary RCS activity concentration of 1.0 μCi/gm DE I-131.

- CIS Case: For this case, the licensee assumes that the RCS transient associated with the SGTR causes an iodine spike in the primary RCS. It is assumed that the iodine release rate from the fuel rods to the primary RCS increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value, which is consistent with the guidance of RG 1.183 when no fuel failure is assumed. Also, consistent with RG 1.183 guidance, the spike duration persists for a period of 8 hours.

In addition to the TS limited iodine concentration in the primary RCS, the licensee also assumes the TS maximum secondary coolant iodine concentration is available for release, as inferred by RG 1.183 guidance. As shown in TS 3.7.14, and as input to the LAR analysis, the maximum secondary coolant iodine concentration at CCNPP is 0.1 μCi/gm DE I-131.

Consistent with RG 1.183 guidance, the licensee assumes that the radioactive iodine speciation released from the SGs is 97.00% elemental and 3.00% organic.

After analyzing these two cases, the licensee determined that the dose consequences from the CIS case are bounded by those of the PIS case, when no fuel failure is assumed. However, because the dose acceptance criteria, as shown in RG 1.183, are lower for the CIS case, the dose consequences from the CIS case are also relevant to this DBA analysis.

3.2.4.2 Transport Methodology and Assumptions

The licensee assumes that the decrease in pressurizer level continues until the pressurizer empties, dropping the primary pressure to hot leg saturation pressure. Then an upper head void will begin to appear and an SIAS will be actuated. High-pressure safety injection (HPSI) flow first contributes to further reduction of the primary pressure due to its cooling effect but eventually injects enough coolant to refill the pressurizer, restore the primary pressure to provide subcooling, and collapse the upper head void. The reactor trip also generates a turbine trip causing the secondary pressure to rapidly increase due to closure of the turbine valve. The licensee assumes that steam bypass valves are not available to mitigate the rise in secondary pressure; however, the licensee does assume that the action of the Atmospheric Dump Valves (ADVs) and Main Steam Safety Valves (MSSVs) will limit the secondary pressure until the operator is able to assume control. The operator identifies the event from the radiation alarms, the increasing radioactivity in the condenser off-gas monitor, SG blowdown monitor, stack gas or main steam line monitors, the reactor trip on low RCS pressure, the decreasing pressurizer level, and the increasing water level in the affected SG.

After the operator identifies the event, the operator initiates a cooldown of the RCS according to CCNPP SGTR emergency operating procedures (EOPs). Specifically, the licensee states that this action is governed by SGTR EOP-6. Cooldown is performed to relieve secondary pressure and stop the cycling of the MSSVs by bringing the primary hot leg temperature down to 515 °F. In the licensee's analysis, the worst-case single failure blocks the ADV of the intact SG at the beginning of the event. Thus, the initial cooldown is carried out using the ADV of the affected SG only. However, the licensee assumes that the operator will take action to unblock the ADV of the intact SG and isolate the affected SG in order to mitigate the release of radioactivity to the environment in accordance with plant procedures. After the operator isolates the affected SG, the operator will continue to cool down the RCS using the intact SG and the affected SG level will be maintained by using backflow to the RCS. At this point, procedures allow for the operator to have three cooldown mode options in order to attain SDC. However, only two of these options rely on safety-related equipment which would be appropriate for mitigating the design basis SGTR accident. The licensee's evaluation of these two options resulted in the following:

- The operator continues the cooldown via the ADV of the unaffected SG until SDC entry conditions are reached. It will take approximately 14 days for the decay heat generation to decline to a level that can be removed via a single SG and ADV. Instead of assuming a 0 - 2-hour cooldown via the ADV of the affected SG, followed by a 2 - 30-day cooldown via the unaffected SG, the licensee conservatively assumes a 0 - 30-day cooldown via the ADV of the unaffected SG to model this mode.

- The operators can re-open the ADV of the affected SG for up to 8 hours after an initial cooldown of 24 hours post-accident. The licensee models this by assuming an initial 0 - 2-hour cooldown via the ADV of the affected SG, followed by a 2 - 24-hour cooldown via the ADV of the unaffected SG, then a 24 - 32-hour cooldown via the ADV of the affected SG.

For the transport activity in their dose analysis, the licensee assumes that the SGTR is initiated with the primary RCS activity and the TS limited secondary coolant activity uniformly distributed throughout their respective coolant loops. For the CIS case, the primary RCS activity is

released homogeneously into the coolant over an 8-hour duration. The iodine and gas in the primary RCS are released at a 200 gpd rate into the intact SG and at the calculated time-dependent tube rupture leak rate into the ruptured SG. From there, the noble gas activity is leaked through the ADVs and MSSVs directly into the environment, when the ADVs and MSSVs are in the open position. The iodine activity, however, is leaked to the ADVs and MSSVs of the ruptured SG, where a percentage is vented directly to the environment via flashing. The remaining iodine is added to the secondary system, where it is then released through the ADVs by steaming, with a partition factor of 100, when the ADVs and MSSVs are in the open position. The licensee also accounts for the release of activity in the secondary coolant by immediately releasing the maximum secondary coolant activity allowed by the TS, 0.1 μCi/gm DE I-131, directly to the environment.

3.2.4.3 Conclusion

For each of the design basis cooldown modes following the postulated SGTR accident, 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 bounding PIS SGTR accident at CCNPP are a TEDE of 25 rem at the EAB for any 2 hours, 25 rem at the outer boundary of the LPZ and 5 rem for access and occupancy of the control room for the duration of the accident. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the CCNPP UFSAR as design bases. The staff also performed an independent calculation of the dose consequences of the SGTR accident using the licensee's assumptions for input to the RADTRAD computer code. The staff's calculation confirmed the licensee's dose results. The major parameters and assumptions used by the licensee and found acceptable to the staff are presented in Table 3.2.4 (attached). The results of the licensee's design basis radiological consequence calculation are provided in Table 3.2 (attached). The EAB, LPZ, and CR doses estimated by the licensee for the SGTR accident were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.2.5 Seized Rotor Event / Locked Rotor Accident (LRA)

The current CCNPP design basis LRA analysis is based on the traditional accident source term described in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." The CCNPP licensing basis analysis is presented in UFSAR Chapter 14.16, "Seized Rotor Event." To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and control room radiological consequences of the postulated LRA. This analysis was performed to demonstrate that the engineered safety features designed to mitigate the radiological consequences at CCNPP will remain adequate after implementation of the AST.

The licensee submitted the AST-based analysis of the LRA as part of the LAR. Included in this analysis are the assumptions, parameters, and newly calculated offsite and control room doses associated with implementing the AST methodology. The licensee cites RG 1.183 as providing the primary radiological analysis assumptions for their analysis of the postulated design basis LRA. Specifically, the NRC staff's guidance is detailed in Appendix G of that document.

The licensee has defined the design basis LRA as a complete seizure of a Reactor Coolant Pump (RCP) shaft. The licensee has postulated the seizure to occur due to a mechanical

failure or a loss of component cooling to the pump shaft seals, and has determined the most limiting seized rotor event to be an instantaneous RCP shaft seizure at hot full power (HFP). The reactor coolant flow through the core would be asymmetrically reduced to three pump flow as the result of a shaft seizure on one pump.

3.2.5.1 Activity Source

To determine the maximum offsite and control room dose resulting from the postulated design basis LRA, the licensee assumed 5.0% fuel cladding failure. The failed fuel fraction is consistent with the current CCNPP design basis as shown in CCNPP UFSAR Chapter 14.16.3.3. The licensee states that the analysis of record calculates that actual fuel failure, resulting from an LRA, would be 1.02%; therefore, the use of 5.0% in the design basis LRA dose consequence analysis is conservative. For the postulated LRA, the initial thermal power is assumed to be 2,754 MWth, which is a factor of 1.02 times the current licensed thermal power of 2,700 MWth, in order to account for the ECCS evaluation uncertainty. An additional pin power peaking factor of 1.70 is applied. The licensee states that the current CCNPP COLR identifies total integrated radial peaking factors of less than, or equal to, 1.65 for both Unit 1 and 2. Therefore, the assumed pin power peaking factor of 1.70 is conservative. Additionally, the licensee accounts for current licensed values for fuel enrichment and burnup when determining the core isotopic inventory. To account for gap fraction uncertainty in fuel that does not meet the criteria specified in footnote 11 of RG 1.183, the licensee multiplied these gap fractions by a factor of two. This is a conservative approach that is acceptable to the staff.

The maximum TS limited primary RCS iodine concentration that is assumed to be available for release, as one of the changes requested by this LAR, is 0.5 µCi/gm DE I-131. This identifies a requested reduction from the current licensing basis activity concentration of 1.0 µCi/gm DE I-131.

In addition to the TS limited iodine concentration in the primary RCS and activity resulting from assumed fractions of failed fuel, the licensee also assumes that the TS maximum secondary coolant iodine concentration is available for release, as inferred with RG 1.183 guidance. As shown in TS 3.7.14, and as input to the LRA analysis, the maximum secondary coolant iodine concentration at CCNPP is 0.1 µCi/gm DE I-131. Consistent with RG 1.183 guidance, the licensee assumes that the radioactive iodine speciation released from failed fuel is 95.00% aerosol (particulate), 4.85% elemental, and 0.15% organic. Whereas, the radioactive iodine speciation released from the SGs is 97.00% elemental and 3.00% organic.

3.2.5.2 Transport Methodology and Assumptions

The primary function of the RCPs is to provide forced coolant flow through the reactor core. The RCS is a two-loop two-SG system with four cold legs. There are four RCPs in the RCS which are located in the SG cold legs. It is assumed that instrumentation will alert the operators to any incipient failures in the RCP motors or seals. Non-reverse rotation devices are provided on the pump motors to prevent the pump from “windmilling” in the reverse direction, to allow starting the pump with 90% rated voltage, and to limit backflow through a stopped pump from thereby bypassing the core.

The postulated LRA at CCNPP is initiated at HFP by an instantaneous complete seizure of a single RCP shaft. With the reduction of core flow due to the loss of an RCP, the core coolant

temperatures will increase. Assuming a positive moderator temperature coefficient (MTC), the core power will increase. The core average heat flux will decrease slightly due to the increasing core temperatures. The insertion of Control Element Assemblies (CEAs), due to a low RCS flow trip, will terminate the power increase; however, a limited number of fuel pins will experience DNB for a short period of time and are assumed to fail. The initial primary RCS activity and gap activity released into the primary RCS from the failed fuel leaks into and combines with the activity in the secondary system. Releases to the environment occur from the SGs, via the ADVs, and from the condenser. The licensee does not, however, credit the condensers for holdup or dilution of any airborne activity release. This is consistent with an assumption of a coincident LOOP.

For the LRA, the licensee assumes constant RCS leakage until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C, as specified by RG 1.183. The licensee assumes an 8-hour cooldown period, based on actual plant operation as shown in design analysis CA06451, Attachment B, which was submitted by the licensee to support their LAR request. Consistent with the CCNPP design basis, the steam release from the first 1800 seconds was calculated using a CESEC[1] thermohydraulic code run, and the steam release from 1800 seconds to 8 hours is based on a simple energy balance methodology. That is, the steam released from 1800 seconds to 8 hours is based on the amount of steam required to remove the residual heat from the primary and secondary systems, the decay heat generated in the core, and the reactor coolant pump heat.

The licensee assumes that the LRA occurs at time equal to 0, releasing the failed fuel gas gap iodine, noble gas, and alkali metal activities into the primary system to mix immediately and homogeneously with the primary RCS. The current licensing basis at CCNPP restricts the primary-to-secondary leakage rate to a TS limited 200 gpd. The noble gas from the gas gap of the failed fuel, and the TS limited concentration of noble gas activity assumed to be in the primary RCS, are released at a rate of 200 gpd into the SGs and then directly through the ADVs into the environment. The alkali metals and iodine from the gas gap of failed fuel, and the TS limited concentration of iodine activity assumed to be in the primary RCS, are also released into the SGs at the TS rate of 200 gpd. A percentage of this activity leakage is vented directly through the ADVs into the environment via flashing. The alkali metal activity and remaining iodine activity is added to the activity in the secondary system and, subsequently, released from the SGs by steaming. For the iodine activity released from the SGs by steaming through the ADVs, a partitioning factor of 100 is used. A partitioning factor of 1, representing no partitioning, is assumed for alkali metal activity released by steaming. The partitioning factor of 100 is consistent with applicable guidance found in RG 1.183, and is, therefore, acceptable. The retention of particulate radionuclides in the SGs is limited by the moisture carryover from the SGs. Therefore, the licensee used a moisture carryover fraction of 0.001 to characterize the flashing and steaming release of alkali metal activity. The licensee assumed that, according to design specifications referenced in their analysis, the CCNPP SGs have a moisture carryover fraction of less than 0.0005. However, the moisture carryover fraction of 0.001 was used in their analysis for conservatism.

[1]The CESEC code is customized to provide thermohydraulic analysis capability for Combustion Engineering nuclear steam supply systems. See ABB Topical Report "CESEC, Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," Enclosure 1-P to LD-82-001, December 1981.

With regard to flashing; the licensee assumed that all alkali metal activity leakage, from the SG volume to the environment, flashes. However, for iodine flashed coolant flow from the SG volume, and for all flashed coolant flow directly from the RCS to the environment, flashing factors were calculated. The licensee calculated the coolant activity flashing fractions based on calculations in the design analysis CA03516, A-CC-FE-0060 Rev. 1, "Calvert Cliffs Units 1 and 2 Seized Rotor Analysis," as specified in the submitted AST LRA analysis. The CA03516 analysis also calculates the 8-hour cooldown methodology from HFP to SDC. Using a constant enthalpy process assumption, the licensee determined the flashing fractions by comparing projected primary system and secondary system coolant temperatures, and associated enthalpies, as a function of time following an accident. These RCS and SG cooldown temperatures as a function of time are shown in Attachment B of the LRA reanalysis included in the LAR submittal. The licensee calculated an 8.84% flashing fraction for the first 15 minutes of the RCS to environment flashed flow, but conservatively used 10% for that period. For times after 15 minutes and for the duration of the release, until SDC is reached at 8 hours, the licensee assumed a 1% flashing fraction, which is a conservatively rounded value from their calculation.

The licensee assumed no credit for cleanup mechanisms (spray, filtration, plateout) in the primary or secondary systems for any releases.

3.2.5.3 Conclusion

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. The accident-specific dose acceptance criteria for the LRA at CCNPP are a TEDE of 2.5 rem at the EAB for any 2 hours, 2.5 rem at the outer boundary of the LPZ and 5 rem for access to and occupancy of the control room. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the CCNPP UFSAR as design bases. The staff also performed an independent calculation of the dose consequences of the LRA using the licensee’s assumptions for input to the RADTRAD computer code. The staff’s calculation confirmed the licensee’s dose results. The major parameters and assumptions used by the licensee and found acceptable to the staff are presented in Table 3.2.5 (attached). The results of the licensee’s design basis radiological consequence calculation are provided in Table 3.2 (attached). The EAB, LPZ, and CR doses estimated by the licensee for the LRA were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.2.6 Control Element Assembly Ejection Accident / Control Rod Ejection Accident (CREA)

The current CCNPP design basis CREA analysis is based on the traditional accident source term described in TID-14844, “Calculation of Distance Factors for Power and Test Reactor Sites.” The CCNPP licensing basis analysis is presented in UFSAR Chapter 14.13, “Control Element Assembly Ejection.” To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and control room radiological consequences of the postulated CREA. The analysis was performed to demonstrate that the ESFs designed to mitigate the radiological consequences at CCNPP will remain adequate after implementation of the AST.

The licensee submitted the AST-based analysis of the CREA as part of the LAR. Included in this analysis are the assumptions, parameters, and newly calculated offsite and control room doses associated with implementing the AST methodology. The licensee cites RG 1.183 as providing the primary radiological analysis assumptions for the analysis of the postulated design-basis CREA. Specifically, the NRC staff's guidance is detailed in Appendix H of that document.

The licensee has defined the design-basis CREA as a rapid, uncontrolled, total withdrawal of a single or dual CEA, where a dual CEA is two CEAs connected to a single CEA extension shaft. This is postulated to occur as a result of a complete instantaneous circumferential rupture of either the Control Element Drive Mechanism (CEDM) pressure housing or the CEDM nozzle from the reactor vessel closure head. The pressure of the RCS is assumed to cause the ejection of the extension shaft through the rupture and the movement of the CEA to a fully-withdrawn position. The most limiting CRDA is assumed to be rapid total withdrawal of the highest worth CEA within 0.05 seconds and the breaching of the RCS pressure boundary. The immediate reactor core response is expected to be an exponential increase in core power. The insertion of the CEAs will terminate the event. The licensee analyzed both the HFP and hot zero power (HZP) cases and determined the HFP case to be more limiting with respect to dose because it results in more failed fuel and higher steaming potential.

The licensee independently analyzed two release paths: (1) containment leakage, and (2) secondary RCS steaming through SG valves. The licensee's design analysis shows that the latter scenario is more limiting with respect to dose. This is likely due to credible activity removal mechanisms in containment, as discussed in the Section 3.2.6.2.1, as well as the difference in total activity released to the environment activity.

3.2.6.1 Activity Source

As shown in CCNPP UFSAR Chapter 14.13, the current CREA analysis assumes that 8% of the fuel will reach incipient centerline melt temperature, and none will experience clad damage. However, to determine the maximum offsite and control room dose resulting from the postulated design-basis CREA, the licensee assumed, in their revised CREA analysis, that in addition to the 8% fuel melt, 2% of the fuel will experience clad damage. The 8% that is assumed to melt will also experience cladding failure. As a basis for the conservatism of this assumption, the licensee cites a CREA analysis performed for CCNPP Unit 2, Cycle 8, and an SE of June 30, 1987, titled "Revised Safety Evaluation Supporting Amendment No. 108 to Facility Operating License No. DPR-69," that predicts and acknowledges that less than 1% of all fuel will reach the incipient fuel melting threshold in the event of a CREA. The licensee also cites CREA analyses for the 12th and 15th Cycles of CCNPP Unit 1, which calculate no fuel melt and 8% fuel melt, respectively, at HFP. For the CREA, the initial thermal power is assumed to be 2,754 MWth, which is a factor of 1.02 times the current licensed thermal power of 2,700 MWth, in order to account for the ECCS evaluation uncertainty. An additional pin power peaking factor of 1.70 is applied. The licensee states that the current CCNPP COLR identifies total integrated radial peaking factors of less than, or equal to, 1.65 for both Unit 1 and 2. Therefore, the assumed pin power peaking factor of 1.70 is considered conservative. Additionally, the licensee accounts for current licensed values for fuel enrichment and burnup when determining the core isotopic inventory. The licensee also accounts for the TS limited primary and secondary RCS activity in the calculation of activity available for release. The TS limited primary RCS iodine concentration that is assumed, as one of the changes requested by this LAR, is 0.5 μCi/gm DE

I-131. This identifies a requested reduction in the current licensing basis activity concentration of 1.0 μCi/gm DE I-131. The assumed secondary RCS activity is the TS limit of 0.1 μCi/gm DE I-131.

The following are the release fractions associated with the two release scenarios analyzed by the licensee:

- Containment Leakage Release: The licensee assumed that 100% of the noble gas and 25% of the iodine contained in the fuel, which is estimated to reach initiation of melting, and 10% of the noble gas and 10% of the iodine, which is contained in the gas gaps of the fuel which experiences clad failure, is released into the primary RCS. The licensee also accounts for other particulate nuclides available for release from melted fuel, per RG 1.183 guidance.

- Secondary RCS Release: The licensee assumed that 100% of the noble gas and 50% of the iodine contained in the fuel, which is estimated to reach initiation of melting, and 10% of the noble gas and 10% of the iodine, which is contained in the gas gaps of the fuel which experiences clad failure, is released into the primary RCS. The licensee also accounts for other particulate nuclides available for release from melted fuel, per RG 1.183 guidance.

3.2.6.2 Transport Methodology and Assumptions

The containment leakage release pathway considers a release of activity from the primary RCS directly into containment where the licensee assumes it mixes instantaneously and homogeneously. The licensee assumes that the activity leaks from the containment atmosphere to the environment at the design basis TS containment leak rate.

3.2.6.2.1 Containment Leakage Release

The current CCNPP design basis containment leak rate, L_a, is equal to 0.2 % per day at containment peak pressure as expressed in CCNPP TS 5.5.16. The licensee proposes to reduce L_a to a value of 0.16 % per day, which is to be used for the first day. Following the first 24 hours, the assumed value for L_a is reduced by half. This is consistent with the guidance of RG 1.183.

The airborne activity in the CCNPP containment following the postulated CREA is mitigated by natural deposition of fission products in aerosol form and removal by the containment IRS. The following subsections discuss the evaluation of the credit that is taken for activity mitigation by these processes.

The use of models for the various mechanisms for iodine removal, when more than one is used simultaneously for the same iodine species in a dose analysis, should consider the effect of one model on the others. Because each model used by the licensee does not account for removal through the other model, the use of both the referenced natural deposition models and the filtration removal system in the same region of containment for the same time period is recognized as potentially non-conservative. Although both natural deposition and the IRS are acting on the overall in-containment aerosol and elemental iodine source term, the total effect from each of these removal mechanisms is not the same as would be found by simply adding

the removal coefficients for each model for a given time period. The modeled removal of particulate and aerosol iodine by natural deposition is far overwhelmed by the removal modeled for the containment IRS and the resulting offsite and control room doses are minimally affected by the inclusion of natural deposition. Therefore, for this LAR, the NRC staff does not deem it necessary for the licensee to recalculate the iodine removal and subsequent resulting CREA dose consequences using a more conservative modeling of iodine removal mechanisms. While the inclusion of the effects of natural deposition in conjunction with containment IRS iodine removal could be non-conservative, the staff's confirmatory analysis indicates that the overall iodine removal as determined by the licensee in this application is conservative and is, therefore, acceptable.

3.2.6.2.1.1 Containment Iodine Removal System (IRS)

The IRS at CCNPP has three filter units, called iodine removal units (IRUs), each with a capacity to handle 50% of the required air flow. Each IRU consists of activated charcoal filters preceded by HEPA filters and has a 20,000 cfm ±10% flowrate. The CCNPP IRS is designed such that one IRU is connected to one of two buses, while the other two IRUs are connected to the other bus. The licensee conservatively assumes that only one IRU is automatically initiated at 63 seconds, accounting for delays associated with a LOOP followed by a worst-case single failure of an EDG. The licensee further assumes that a second IRU is manually initiated at 20 minutes. Therefore, at 63 seconds the filtration efficiencies credited are 45% for aerosol and elemental, and 15% for organic iodine forms, and after 20 minutes these efficiencies are increased to 90% for aerosol and elemental, and 30% for organic. The licensee's treatment of the IRS IRU initiation is conservative and consistent with current CCNPP design basis assumptions, as shown in CCNPP UFSAR 14.24.3 and is, therefore, acceptable to the NRC staff.

3.2.6.2.1.2 Natural Deposition

The licensee's analysis assumed removal of airborne activity in aerosol form by natural deposition in containment following the postulated CREA using Powers' simplified natural deposition model in the dose consequences computer code described in NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," and its supplements. Powers' simplified natural deposition model is described in NUREG/CR-6189, "A Simplified Model of Aerosol Removal By Natural Processes in Reactor Containments." The licensee used the 10th percentile confidence interval (90 percent probability) removal values implemented in the RADTRAD code. The Powers natural deposition model was derived by correlation of results of Monte Carlo uncertainty analyses of detailed models of aerosol behavior in the containment under accident conditions. The licensee has taken no credit for removal of elemental or organic iodine species by natural deposition. The licensee's use of the Powers model, as implemented in the NRC computer code, RADTRAD, and as discussed in RG 1.183, is acceptable to the NRC staff.

3.2.6.2.2 Secondary RCS Release

The secondary RCS release pathway considers a release of activity from the primary RCS directly into secondary RCS via SG tube leakage at the CCNPP TS primary-to-secondary leak rate of 200 gpd. The licensee assumed that the activity leaks to the environment through the SGs via the ADVs and MSSVs at calculated post-accident steaming rates. Consistent with

RG 1.183 guidance, the licensee assumes that the radioactive iodine speciation released from failed fuel is 95.00% aerosol (particulate), 4.85% elemental, and 0.15% organic, whereas, the radioactive iodine speciation released from the SGs is 97.00% elemental and 3.00% organic.

For this release pathway, the licensee assumed the duration of the cooldown from HFP to SDC to be based on the CCNPP TS maximum cooldown rate of 100 °F/hour. HFP is defined as 574.5 °F and 2250 psia, per CCNPP UFSAR Table 4.1 and Figure 4.9, and SDC is defined as 300 °F and 270 psia, per the plant Emergency Operating Procedures. Therefore, as analyzed, this CREA scenario assumed an 8-hour cooldown duration that is based on actual plant operation.

The licensee developed the methodology for the steaming associated with the 8-hour cooldown from the LRA analysis. The licensee states, and the NRC staff agrees, that the LRA cooldown methodology is applicable to the CREA with minor corrections. Consistent with the CCNPP design basis, the steam release from the first 1800 seconds was calculated using a CESEC thermohydraulic code run, and the steam release from 1800 seconds to 8 hours is based on a simple energy balance methodology. That is, the steam released from 1800 seconds to 8 hours is based on the amount of steam required to remove the residual heat from the primary and secondary systems, the decay heat generated in the core, and the reactor coolant pump heat. To evaluate iodine releases, the steam release rates are divided by a partition factor of 100, as described in RG 1.183, and used as input to the RADTRAD code for calculation of dose.

The licensee used the same treatment as discussed for the LRA analysis in Section 3.2.5.2 to determine the 10% and 1% flashing, which are respectively applied from 0 to 15 minutes and 15 minutes to 8 hours following the accident initiation.

3.2.6.3 Conclusion

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. The accident-specific dose acceptance criteria for the CREA at CCNPP are a TEDE of 6.3 rem at the EAB for any 2 hours, 6.3 rem at the outer boundary of the LPZ and 5 rem for access to and occupancy of the control room. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the CCNPP UFSAR as design bases. The staff also performed an independent calculation of the dose consequences of the CREA using the licensee's assumptions for input to the RADTRAD computer code. The staff's calculation confirmed the licensee's dose results. The major parameters and assumptions used by the licensee and found acceptable to the staff are presented in Table 3.2.6 (attached). The results of the licensee's design basis radiological consequence calculation are provided in Table 3.2 (attached). The EAB, LPZ, and CR doses estimated by the licensee for the CREA were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.3 Control Room Habitability and Modeling

The current CCNPP DBA analysis, as shown in UFSAR Chapter 14, does not calculate control room dose. Therefore, the control room dose model provided in the revised DBA accident analyses that support this AST-based LAR, represents a change in the CCNPP licensing basis.

For their revised analyses, the licensee assumes a nominal control room emergency ventilation system recirculation flow of 10,000 cfm ±10%, and credits a 90% filtration efficiency for elemental and organic iodine, and a 99% filtration efficiency for particulate iodine. For conservatism, the lower flow uncertainty value of 9000 cfm is used for modeling. The system filtration is assumed to initiate after 20 minutes while the recirculation flow is assumed to be present from the accident onset. The licensee also credited installation of automatic isolation dampers and radiation monitors on the Auxiliary Building Roof. These dampers are credited for limiting activity ingress into the control room from either the West Road Inlet or the Turbine Building, thereby limiting the atmospheric dispersion coefficient value.

In response to GL 2003-01, the licensee's letter dated December 5, 2003 (ML033440342), indicates that the measured unfiltered inleakage into the CCNPP control room is 3000 cfm ± 250 cfm. For the DBA analyses, the licensee assumes that all makeup flow, or intake, to the CCNPP control room is unfiltered at 3500 cfm. This value is conservative and provides margin for future measurements of control room inleakage.

3.4 Equipment Qualification

The proposed amendment will revise the accident source term in the design basis radiological consequence analyses in accordance with 10 CFR 50.67. The proposed accident source term revision replaces the current methodology that is based on TID - 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," with the AST methodology described in RG 1.183. This LAR is for full implementation of the AST with the exception that the current methodology of TID-14844 will continue to be used as the radiation dose basis for equipment qualification (EQ).

In SECY-99-240, "Final Amendments to 10 CFR Part 21, 50, and 54 and Availability For Public Comment of Draft Regulatory Guide DG-1081 and Draft Standard Review Plant Section 15.0.1 Regarding Use of Alternate Source Terms at Operating Reactors," (ADAMS accession No. ML993080146) the NRC staff considered the potential impact of the postulated cesium concentration on the operability of safety systems at current operating reactors. Staff analyses have shown that the EQ doses determined using the current TID-14844 source term are more limiting than those calculated using the NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," (ADAMS accession No. ML041040063) source term for exposure periods less than 30 days to 4 months following the accident. The postulated increase in the cesium concentration is not a concern for those systems and components having a safety function that is performed and completed earlier than 30 days following an accident. The staff concludes that continued plant operation does not pose a threat to public health and safety since this equipment will remain capable of performing its intended design functions. The small increase in long-term dose is typically compensated by the significant conservatism incorporated in EQ dose assessments. This conservatism includes: application of the worst dose in a zone to most components in the zone, assumption that all piping that could contain post-accident activity doses contain such activity, and neglecting the shield effects of intervening equipment. In addition, significant margin typically exists between a component's calculated dose and the component's qualification dose. For most components, there is also additional margin between the qualification dose and the dose that would cause component failure. In RG 1.183, the staff specified that a licensee making an AST submittal can continue to use the TID-14844 source term for EQ.

Based on the above, the NRC staff concurs with the licensee's statement that TID-14844 continue to be used as the radiation dose basis for equipment qualification. Current EQ doses based on TID-14844 source term provide adequate justification for the continued operability of EQ components and is acceptable to the staff.

3.5 Exceptions to RG 1.183

RG 1.183 provides guidance to licensees of operating power reactors on implementation of alternative source terms. The licensee followed the guidance provided in RG 1.183 with the exception of using the fuel gap fractions in Table 3.

The licensee data base indicated that linear heat generation rates in excess of 6.3 kw/ft exist for fuel assemblies with burnup in excess of 54 GWd/MTU at both units. Therefore, since the applicable criteria in RG 1.183 would not be met, the licensee did not use the non-LOCA gap fractions in Table 3. As an alternative, pursuant to RG 1.183, fission gas release calculations using NRC-approved methodologies may be considered on a case-by-case basis. To be acceptable, the licensee's calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specified fuel load.

The NRC staff reviewed the methodology used to determine the core inventory and the gap fractions used for LOCA and non-LOCA transients and accidents, including the FHA. The licensee stated that the ORIGIN-S methodology was used to determine the core inventory. The staff finds the use of the ORIGIN code acceptable in accordance with Section 3.1 of RG 1.183.

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

For the FHA and non-LOCA transients and accidents, the licensee determined the number of fuel assemblies that did not meet the applicability criteria. Additionally, the power history for the most limiting assembly was determined and a bounding peaking factor was determined to obtain the limiting gas gap fraction per ANSI/ANS-5.4-1982. The results of the analysis show that gas gap fractions are in excess of the limits in Table 3, but less than twice the limiting value. Therefore, the licensee increased the fuel/clad gap fission product inventories in Table 3 of RG 1.183 by a factor of 2.0. The factor of 2.0 was used to offset the fact that some fuel assemblies would exceed rod power/burnup criteria in RG 1.183 (See Footnote 11). The factor of 2.0 was applied to all assemblies.

For the FHA, the analysis assumes that gas gap activity from 176 fuel rods of the highest power assembly is released. In addition, the cycle-maximum radial peaking factor is conservatively applied to the fuel rods in the limiting, high burnup assembly which creates a conservative composite worst-case fuel configuration (high power with high gap inventory). Based on these conservative assumptions and the calculations provided in the application, the NRC staff finds a factor of 2.0 applied to the Table 3 gap fractions acceptable.

For the CEA Ejection Analysis, the licensee assumed 10% of the core inventory of noble gases and iodines are present in the fuel/clad gap. This assumption is consistent with RG 1.183 and the staff finds it acceptable.

For the other Non-LOCA events which experience fuel damage (e.g. locked rotor), increasing the RG 1.183 fuel/clad gap fission-product inventory by a factor of 2.0 is conservative because on a core-wide basis, only a small fraction of the fuel rods exceed the applicability criteria. Further, during core-wide transients, the high power, low burnup rods (containing less gap inventory) are more prone to clad failure than the low power, high burnup rods (containing more gap inventory). In addition, the licensee stated that the cycle-maximum radial peaking factor was applied to all failed fuel rods. Based on these conservative assumptions and the calculation provided in the application, the NRC staff finds a factor of 2.0 applied to the Table 3 gap fractions acceptable.

Based upon the above, the NRC staff finds the determination of pressurized-water reactor (PWR) core inventory and the use of the gap fractions for LOCA and non-LOCA transients and accidents acceptable.

3.6 Technical Specification Changes

The proposed license amendment would revise the following TSs that are associated with the analyses performed to support the AST:

3.6.1 Table of Contents

The proposed change deletes reference to TS 3.7.10, “Emergency Core Cooling System (ECCS) Pump Room Exhaust Filtration System (PREFS),” from the Table of Contents.

The licensee requested that the Table of Contents be modified to delete reference to Section 3.7.10, “Emergency Core Cooling System (ECCS) Pump Room Exhaust Filtration System (PREFS).” This is considered an editorial change and is acceptable to the NRC staff because, as discussed in item 3.6.5 below, the staff agreed to delete this TS requirement.

3.6.2 TS Section 1.1, “Definitions”

The proposed change revises the definition of DOSE EQUIVALENT I-131 in TS Section 1.1 to reference Federal Guidance Report 11, ORNL, 1988, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” as the source of thyroid dose conversion factors.

The licensee proposed to revise the definition of DOSE EQUIVALENT I-131 in TS Section 1.1 to reference Federal Guidance Report (FGR) 11, ORNL, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," as the source of thyroid dose conversion factors instead of the current TID-14844 inhalation dose conversion factors. The existing definition is based on the thyroid dose conversion factors provided in specified tables in either Atomic Energy Commission (AEC) Technical Information Document TID-14844, “Calculation of Distance Factors for Power and Test Reactor Sites,” RG 1.109, Rev. 1, “Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I,” or International Commission on Radiation Protection Publication 30 (ICRP 30), “Limits for Intakes of Radionuclides by Workers.” Per RG 1.183, Sections 4.1.1 and 4.1.2, the licensee’s AST DBA analyses, described in Section 3.2 use the thyroid conversion factors listed in Table 2.1 of FGR 11, where applicable. Thus, this proposed revision to the definition of DE 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. The new citations are as cited in RG 1.183 and are, therefore, acceptable. Therefore, the NRC staff finds the proposed revision to the TS 1.1 definition DOSE EQUIVALENT I-131 acceptable.

3.6.3 TS Section 1.1, “Definitions”

The definition of L_a used in the Containment Leakage Rate Testing Program (TS 5.5.16) is being changed. L_a will be reduced from 0.20 percent of containment air weight per day at P_a to 0.16 percent of containment weight per day at P_a.

The NRC staff finds this acceptable because it is a reduction in the previous leakage rate, which is conservative, and it is the value used in the radiological consequence design basis calculations which was found to be acceptable.

3.6.4 TS Section 3.4.15, “RCS Specific Activity”

The limit for RCS activity will be reduced from 1.0 μCi/gm to 0.5 μCi/gm. Accordingly, the dose equivalent I-131 primary coolant specific activity limits on TS Figure 3.14-15-1 were revised.

The licensee proposed to reduce the limit for RCS activity in TS 3.4.15 "RCS Specific Activity" from 1.0 μCi/gm DE I-131 to 0.5 μCi/gm DE I-131. The licensee states that this revision is required to meet control room dose regulatory limits for the design basis SGTR accident, as described in Section 3.2.4. The licensee also uses this revised RCS specific activity limit in the LOCA, CREA, MSLB accident, and LRA analyses, but indicates that this revision is not necessary to meet the regulatory requirements for these accidents. This revision will implement a limit that is more conservative than the existing requirement. The licensee states that an examination of CCNPP Unit 2 Cycles 13 and 14, and Unit 1 Cycles 15 and 16, DE I-131 indicates that the DE I-131 remains well below the proposed revised limits. Also, for Unit 1 and Unit 2, the equilibrium DE I-131 values are below 0.02 μCi/gm. Therefore, the NRC staff finds the proposed revision to CCNPP TS 3.4.15 “RCS Specific Activity” limit to be acceptable because the revised accident analyses continue to meet regulatory limits.

3.6.5 TS Section 3.7.10, “Emergency Core Cooling System (ECCS) Pump Room Exhaust filtration System (PREFS)”

The proposed revision would delete TS 3.7.10.

The proposed revision deletes the LCO, Action, and the associated Surveillance Requirements (SRs) in TS 3.7.10, ECCS PREFS, and deletes references to PREFS filter testing in TS 5.5.11, “Ventilation Filter Testing Program.”

The ECCS PREFS filters air from the area of the active ECCS components during the recirculation phase of a LOCA. The ECCS PREFS consists of two independent and redundant fans, a prefilter, a HEPA filter, and an activated charcoal adsorber section for removal of gaseous activity (principally iodines). Ductwork, valves or dampers, and instrumentation also form part of the system.

The ECCS pump rooms for Units 1 and 2 are served by the common waste processing area ventilation supply system. The ECCS PREFS operates during normal unit operations. The ECCS pump room exhausts may be directed through HEPA filters prior to emptying into the

main plant vent. When the ECCS pumps are operated post-accident, air flow from the ECCS pump room area will be diverted through the charcoal filters by manual remote actuation in the Control Room. However, the operation of this system and the resultant effects on offsite dose calculations are not credited in the accident analysis. This system provides defense-in-depth only. Fan-coil coolers are installed in each ECCS pump room to provide additional cooling, if necessary, during pump operation.

The ECCS PREFS may be used for normal, as well as post-accident, atmospheric cleanup functions. In MODES 1, 2, 3, and 4, the ECCS PREFS is required to be OPERABLE consistent with the OPERABILITY requirements of the ECCS. In MODES 5 and 6, the ECCS PREFS is not required to be OPERABLE, since the ECCS is not required to be OPERABLE.

The licensee's offsite and control room dose calculations do not assume any leakage from the ECCS pump room. This assumption is part of the existing licensing basis and will not change upon implementation of the AST. While operation of the PREFS is not credited in the accident analyses, the PREFS is the only ventilation system supporting the ECCS pump room and, as such, the PREFS provides defense-in-depth following a postulated accident.

The NRC staff was concerned that eliminating TS 3.7.10 would result in the elimination of a safety-related ventilation system that was fully capable of providing a backup capability to reduce both offsite and control room dose. The NRC staff asked the licensee to address its assumption of zero leakage from the ECCS pump room during post-accident conditions and to provide assurance that any potential leakage from pump seals and valves in the ECCS pump room would be minimal and would not be expected to provide any consequential contribution to the offsite and control room dose calculations.

By letter dated July 17, 2007, the licensee responded to the NRC staff's concern emphasizing that (1) the ECCS pumps have mechanical seals that minimize seal leakage, (2) periodic integrated leak tests of the ECCS systems required by TS 5.5.2, "Primary Coolant Sources Outside Containment," have demonstrated minimal observed system leakage, (3) immediate actions are taken to eliminate or minimize any leakage observed during testing required by TS 5.5.2, and (4) a modification to the PREVS fan circuitry will prevent automatic post-accident system startup thus preventing any motive force to circulate any radioactive leakage outside of the ECCS pump room.

The HPSI pumps, low pressure safety injection (LPSI) pumps, and the containment spray pumps located in the ECCS pump room are all equipped with mechanical seals. The mechanical seals are the most likely source of pump leakage during pump operation and the most likely failure mechanism is the failure of the mechanical seals to seat and/or seal upon pump startup. TS 5.5.2 requires that an integrated leak test be conducted at least once every 24 months for each system outside containment that could contain highly radioactive fluids following an accident or transient. The HPSI system, LPSI system, and the containment spray system components located in the ECCS pump room are included in the testing required by TS 5.5.2. The integrated leak test consists of filling the system with water and operating a system pump for 10 minutes. The 10-minute test provides adequate assurance that the mechanical seals will seat and seal properly upon pump startup.

Following the integrated leak test pursuant to TS 5.5.2, the system is inspected for leaks by personnel qualified for VT-2 visual examination. When a new leak is identified, it is entered into

the licensee's corrective action program and evaluated in accordance with the licensee's fluid leak management program. The fluid leak management program was developed to ensure external leakage from plant systems is classified, prioritized and managed or corrected in a timely manner. The goal is to have a zero tolerance of detrimental external leakage. The safety injection systems in the ECCS pump rooms are classified at the highest level of leak tightness. The licensee provided the results of integrated leak tests performed from 1997 to 2005. Based upon the results in the table below, the NRC staff concludes that the systems have been maintained reasonably leak-tight.

Test Year	Unit 1 Results	Unit 2 Results
2005	No active leakage	No test results located
2003	LPSI leak-off line plug leak - 2 drops/min	No active leakage
2001	HPSI mechanical seal leak - no leak rate recorded	HPSI threaded connection - 6 drops/min
1999	No active leakage	No active leakage
1997	No active leakage	Containment spray mechanical seal - 20 drops/min

Finally, the licensee provided the following response in their letter dated July 17, 2007, that describes the circuitry modification to prohibit any motive force from circulating any system leakage outside the ECCS pump room.

> During the recirculation phase following an accident, radioactive sump water is recirculated through the ECCS pumps and could leak through various valves and reach the refueling water tank (RWT). Thus, there is a potential for an unmonitored release pathway through the RWT, which is vented directly to the atmosphere. This source is accounted for in the submitted design basis LOCA analyses. However, due to the implementation of the leakage reduction program in accordance with TMI Action Item III.D.1.1, no leakage is assumed from the ECCS through the ECCS pump room to the Auxiliary Building. However, if there were any fluid leakage into the ECCS Pump Room, any airborne radioactivity in the ECCS Pump Room could be transported to a filtered (HEPA filter), monitored release path (the Auxiliary Building vent stack). This release path is evaluated in the accident dose analysis (but does not assume a release from the ECCS Pump Room). To ensure that any effluent from this release path would have less effect on the Control Room dose than an equivalent effluent from the RWT, we are proposing to modify the ECCS Pump Room ventilation fan control circuit to ensure that the fans do not automatically operate following an accident. When the fan is not operating, gravity dampers in the ventilation ducts close. This eliminates the path from the ECCS Pump Room to the Auxiliary Building vent stack. Without ventilation fan operation, any leakage in the ECCS Pump Room will have no motive force to move it outside of that room. Note that the ECCS Pump Room ventilation system does not provide a cooling function for the equipment in the ECCS Pump Room. The air temperature in the ECCS Pump Room is maintained within limits by the ECCS Pump Room air cooler. Saltwater-cooled fan coil coolers are installed in each ECCS Pump

Room to provide room cooling, if necessary, following an accident. These fan coil coolers are contained within the ECCS Pump Room and not connected to the Auxiliary Building ventilation system.

The licensee assumes that the ECCS pump room, which houses ECCS pumps and valves, does not leak activity to the environment and, therefore, does not include a contribution from flashed activity in this room to the total onsite or offsite dose. Though it is expected that pumps and valves in the ECCS pump room have potential for leakage, the NRC staff agrees that such expected leakage would offer only negligible contributions to the total onsite and offsite post design-basis LOCA dose. This expectation is based on the characteristics of the ECCS pump room which has 2-foot thick concrete walls and water-tight doors. Additionally, the licensee has proposed to modify the ECCS pump room ventilation fan control circuit to ensure that fans do not automatically operate following the initiation of the design-basis LOCA. The NRC staff agrees that this would prevent any potential system leakage from leaving the ECCS pump room.

The NRC staff's assessment of the information provided concludes that: (1) the mechanical seals on the ECCS pumps have contributed to minimal system leakage, (2) the integrated leak tests performed pursuant to TS 5.5.2 have demonstrated minimal system leakage, (3) the licensee has established procedures for identifying and mitigating system leakage, and (4) modifying the PREFS circuitry to prevent automatic startup following an accident will preclude a motive force to transport any system leakage outside of the ECCS pump room. Therefore, the NRC staff concludes that the licensee's assumption of zero leakage from the ECCS pump room is reasonable and it is acceptable for the licensee to delete TS 3.7.10, ECCS PREFS, and delete references to PREFS filter testing in TS 5.5.11, "Ventilation Filter Testing Program."

3.6.6 TS Section 3.7.11, "Spent Fuel Pool Exhaust Ventilation System (SFPEVS)"

In LCO Action a, remove the inoperable conditions involving SFPEVS charcoal adsorber and delete the corresponding surveillance requirement for filter testing (SR 3.7.11.2)

The licensee's proposed revision removes the inoperable conditions associated with the ACTIONS of TS 3.7.11 that includes the spent fuel pool exhaust ventilation system (SFPEVS) carbon adsorber, deletes SR 3.7.11.2 regarding SFPVS filter testing, and deletes references to SFPEVS filter testing in TS 5.5.11 "Ventilation Filter Testing Program." The licensee stated that "Spent Fuel Pool (SFP) carbon and high efficiency particulate air filters are not credited in the proposed revision to the licensing basis to implement the AST;" and that the SFPEVS exhaust fan is the only component credited in the design-basis FHA analysis. As a result, the carbon adsorber requirements contained in this TS do not meet any of the criteria for items for which TS LCOs must be established. The NRC staff's review found this requested change acceptable because, as a result of the adoption of the AST, this TS no longer meets the criteria of 10 CFR 50.36(C)(2)(ii) for inclusion in TSs.

For SR 3.7.11.3, the licensee proposed changing the SR from "Verify each SFPEVS fan can maintain a measurable negative pressure with respect to atmospheric pressure" to " Verify each SFPEVS fan can maintain a measurable negative pressure with respect to adjacent areas." The licensee states that this change will insure that airflow following an accident is into the SFP area. The NRC staff agrees with the licensee and, therefore, finds this change acceptable.

3.6.7 TS Section 3.9.3, "Containment Penetration"

In accordance with Technical Specification Task Force (TSTF)-312, "Administratively Control Containment Penetrations," a note is added to LCO 3.9.3 allowing penetration flow path(s) that have direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative control.

TSTF-312 added a note to LCO 3.9.3 permitting licensees to unisolate penetration flow paths that provide direct access from the containment atmosphere to the outside atmosphere under administrative controls. TS 3.9.3, "Containment Penetrations," is only applicable during CORE ALTERATIONS or during movement of irradiated fuel assemblies within containment. As discussed in TSTF-312, TS 3.6.3, "Containment Isolation Valves," contains a note allowing penetration flow paths to be unisolated intermittently during MODES 1 through 4 under administrative controls. During the shutdown conditions of TS 3.9.3, it was pointed out that opening containment isolation valves is less risk significant than in MODES 1 through 4.

The NRC staff has previously reviewed and approved TSTF-312. The note has been incorporated into the staff's Standard Technical Specifications (i.e., NUREG-1430 through 1434). Since the staff has previously reviewed and approved TSTF-312 and the licensee has established appropriate administrative controls to effectively close the unisolated penetrations if necessary, the staff finds the licensee's proposal acceptable.

3.6.8 TS Section 5.5.11, "Ventilation Filter Testing Program"

The Control Room Emergency Ventilation System (CREVS) flow rate is changed from 2,000 cfm to 10,000 cfm in Sections 5.5.11(a), 5.5.11(b), and 5.5.11(d). The testing requirement for the ECCS PREFS and the SFPEVS are deleted from Sections 5.5.11(a), 5.5.11(b), 5.5.11(c), and 5.5.11(d).

The proposed revision increases the CREVS flow rate from 2,000 to 10,000 cfm in TS 5.5.11, "Ventilation Filter Testing Program." This revision is required to meet the AST control room dose regulatory limits for DBAs. The licensee stated that a plant modification will be performed to increase the CREVS flow rate, which meets the guidance of RG 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and GL 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal." The NRC staff finds this change acceptable, but notes that with the increased flow rate, the proper residence time must be maintained in accordance with the guidance of RG 1.52 and GL 99-02.

In addition, to insure appropriate filtration of the control room air, the licensee has committed to install an additional HEPA filter in the control room filtration system. The 10,000 cfm filtration system will now consist of two HEPA filters in series with a carbon adsorber. The additional HEPA filter to the planned system requires a change to the CREVS pressure drop criteria found in TS 5.5.11d. As a result of the additional HEPA filter, the licensee requested an increase to the allowed pressure drop across the CREVS filtration system from 4 inches water gauge (inwg) to 6 inwg to account for the design change from one HEPA in the filtration system to two HEPAs in the filtration system. The NRC staff's assessment found this to be a reasonable increase in pressure drop across the system based on engineering judgment and, therefore, found this requested change acceptable.

3.6.9 TS Section 5.5.16, “Containment Leakage Rate Testing Program”

The maximum allowable containment leakage rate L_a contained in TS 5.5.16, “Containment Leakage Rate Testing Program” is reduced from 0.20 percent of containment air weight per day at P_a to 0.16 percent of containment air weight per day at P_a.

The maximum allowable containment leakage rate L_a contained in TS 5.5.16, “Containment Leakage Rate Testing Program” is reduced from 0.20 percent of containment air weight per day at P_a to 0.16 percent of containment air weight per day at P_a. This is the same change discussed in item 3.6.3 above and the NRC staff finds this acceptable for the reasons previously cited.

3.6.10 TS Section 5.5.11, “Ventilation Filter Testing Program”

The methyl iodide penetration test percentages for the CREVS, PREVS, and IRS are being revised.

By letter dated July 17, 2007, the licensee proposed changing the Ventilation Filter Testing Program test criteria for both the carbon adsorbers and the HEPA filters. For the carbon adsorber, the licensee proposed changing the methyl iodide penetration test percentage for the CREVS from 5 percent to 4.5 percent. The licensee states that this change supports the analyses assumption of 90 percent efficiency for the carbon adsorber. A 4.5 percent penetration limit and a safety factor of two results in an assumed efficiency of 91 percent. With an assumed 1 percent penetration and system bypass, the credited efficiency will be 90 percent. The NRC staff finds this acceptable because it is consistent with the analyses for the CREVS provided in letters dated November 3, 2005, and March 22, 2007. The safety factor of two is also consistent with the guidance of GL 99-02.

The licensee also proposed changing the methyl iodide penetration percentage for the PREVS and the IRS from 35 percent to 34.5 percent. This change allows a 1 percent penetration and system bypass for the PREVS and IRS while maintaining the efficiencies assumed in the accident analyses. The NRC staff’s assessment finds this acceptable because it is consistent with the design basis LOCA analysis.

For the HEPA filter, the licensee proposed changing the CREVS HEPA filter bank penetration and system bypass from 1 percent to 0.05 percent. This allows an assumed efficiency of 99 percent for the CREVS HEPA. The NRC staff finds this acceptable because it is consistent with the licensee’s analyses provided by letters dated November 3, 2005, and March 22, 2007. It is also consistent with the guidance of RG 1.52, Revision 2.

3.7 Summary

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 analyses with the proposed TS changes. The staff finds 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 2.0. The staff also finds, with reasonable assurance, that the licensee’s estimates of the Control Room, EAB, and LPZ doses will comply with these criteria. The staff

further finds reasonable assurance that CCNPP, 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 DBAs.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change SRs. The NRC staff has determined that the amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (71 FR 2589). Accordingly, the amendments meet 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 amendments.

6.0 CONCLUSION

The staff has reviewed the AST implementation proposed by the licensee for CCNPP, Unit Nos. 1 and 2. The staff also reviewed the plant modifications associated with this proposed implementation. In performing this review, the staff relied upon information placed on the docket by the licensee, staff experience in performing similar reviews and, where deemed necessary, on staff confirmatory calculations. The staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed plant modifications in the context of the proposed AST. The staff finds that the licensee used analysis methods and assumptions consistent with the conservative guidance of RG-1.183, with the exceptions discussed and accepted earlier in this SER. The staff finds the methods and assumptions used by the licensee to be in compliance with applicable requirements. The staff compared the doses estimated by the licensee to the applicable acceptance criteria and to the results estimated by the staff in its confirmatory calculations. The staff finds with reasonable assurance that the licensee's estimates of the total effective dose equivalent due to design basis accidents will comply with the requirements of 10 CFR 50.67 and the guidance of RG-1.183.

The staff finds reasonable assurance that CCNPP, Unit Nos. 1 and 2, as modified by this proposal, will continue to provide sufficient safety margins with adequate defense in depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameters. The staff concludes that the proposed AST implementation and the associated plant modifications are acceptable.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the CCNPP's design basis is superseded by the AST proposed by the licensee. The previous offsite and control room accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR 50.67 or small fractions thereof, as defined in RG-1.183. All future radiological analyses performed to demonstrate compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as described in the CCNPP design basis.

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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Aleem Boatright
Robert Dennig
Natasha Greene
Christopher Brown
Duc Nguyen

Date: August 29, 2007

Table 3.1.2

Unit 1 and 2 Control Room Atmospheric Dispersion Factors

Design Basis Accident	Source / Receptor	χ/Q Values 0-2 Hours sec/m^3	2-8 Hours sec/m^3	8-24 Hours sec/m^3	24-96 Hours sec/m^3	96-720 Hours sec/m^3
MHA	RWT1 / WR	2.57E-3	2.13E-3	8.50E-4	5.71E-4	4.85E-4
FHA	VS2 / TB	1.68E-3	1.34E-3	5.14E-4	3.84E-4	3.12E-4
MSLB	MSG2 / TB	3.48E-3	2.97E-3	1.21E-3	9.22E-4	7.41E-4
SGTR	ADV2 / TB	3.83E-3	3.25E-3	1.32E-3	9.92E-4	7.92E-4
SRE	ADV2 / TB	3.83E-3	3.25E-3	1.32E-3	9.92E-4	7.92E-4
CEAEA	ADV2 / TB	3.83E-3	3.25E-3	1.32E-3	9.92E-4	7.92E-4

Notes: 1. Acronyms under Source/Receptor column stand for: RWT1=Unit 1 Refueling Water Tank; WR=West Road of the Auxiliary Building; VS2=Unit 2 Ventilation Stack; TB=Turbine Building; MSG2=Unit 2 Main Steam Gooseneck; ADV2=Unit 2 Atmospheric Dump Valve

2. The licensee applied the worst-case χ/Q value for each of the source/receptor combinations evaluated to both units, irrespective of the unit for which it was actually calculated.

Table 3.1.3

<u>Unit 1 and 2 Offsite Atmospheric Dispersion Factors</u>

Offsite Dose Location	**χ/Q Values** **0-2 Hours sec/m^3**	**2-24 Hours sec/m^3**	**24-720 Hours sec/m^3**
EAB (1150 meters)	1.44E-4	----	----
LPZ (3219 meters)	3.39E-5	2.20E-6	5.40E-7

Notes: 1. The licensee applied the worst-case χ/Q value to both units, irrespective of the unit for which it was actually calculated. These χ/Q values were applied to each postulated DBA.

2. The EAB value of 1.44E-4 was determined by adjusting the containment release χ/Q value of 1.30E-4 sec/m^3 at the EAB via the Gifford wake model for a ventilation release.

3. The EAB value of 3.39E-5 was determined by adjusting the containment release χ/Q value of 3.30E-5 sec/m^3 at the outer boundary of the LPZ via the Gifford wake model for a ventilation release.

Table 3.2

Licensee Calculated Radiological Consequences of Design Basis Accidents

Design Basis Accident	Control Room Total Dose (rem TEDE)	Control Room Acceptance Criteria (rem TEDE)	EAB Total Dose (rem TEDE)	EAB Acceptance Criteria (rem TEDE)	LPZ Total Dose (rem TEDE)	LPZ Acceptance Criteria (rem TEDE)
LOCA	4.57E+00	*5.0*	1.85E+00	*25*	4.60E-01	*25*
FHA	3.85E+00	*5.0*	1.11E+00	*6.3*	2.62E-01	*6.3*
MSLB						
Failed Fuel	4.63E+00	*5.0*	2.18E-01	*25*	5.77E-02	*25*
CIS	2.08E-01	*5.0*	2.25E-03	*2.5*	1.05E-03	*2.5*
SGTR						
PIS	4.17E+00	*5.0*	4.91E-01	*25*	1.16E-01	*25*
CIS	1.71E+00	*5.0*	1.96E-01	*2.5*	4.84E-02	*2.5*
LRA	7.89E-01	*5.0*	4.10E-02	*2.5*	9.50E-03	*2.5*
CREA	4.76E+00	*5.0*	3.28E-01	*6.3*	8.82E-02	*6.3*

Notes 1. The licensee calculated the EAB dose for the worst 2-hour period of the accident duration.

2. The licensee's Total Dose results have been rounded to three significant digits.

Table 3.2.1

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

Parameter	Value
Reactor Core Power, MWth	2754
Containment Volume, ft^3 Sprayed Unsprayed Total	 1,446,000 543,000 1,989,000
Spray Delay time, min	10
Spray Iodine Removal Coefficient, hr^{-1} Elemental Organic Aerosol/Particulate	 14.82 Not credited. 3.414
Sprayed and Unsprayed Region Mixing Rate, cfm	110,000
Primary Containment Leakage Rate, weight % per day 0 to 24 hrs 24 hrs to 30 days	 0.16 0.08
Aerosol Iodine Natural Deposition Model	Powers 10^{th} Percentile
Containment IRS Flow Rates, cfm 63 seconds to 20 minutes 20 minutes to 720 hours	 20,000 ± 10% 40,000 ± 10%
Containment IRS Activity Removal, % Aerosol/Particulate 63 sec to 20 min 20 min to 30 days Elemental 63 sec to 20 min 20 min to 30 days Organic 63 sec to 20 min 20 min to 30 days	 45 90 15 30 15 30
ECCS Leakage to RWT, cm^3/hr	2000
ECCS Leakage Flashing, %	10
ECCS Iodine Release Species, % Elemental Organic	 97 3
RWT Vent Rate, cfm	4.2
PREVS Filter Efficiency, % Aerosol/Particulate Elemental Organic	 90 90 30
Atmospheric Dispersion Factors	Tables 3.1.2 and 3.1.3

Table 3.2.2

Key Parameters Used in Radiological Consequence Analysis of Fuel Handling Accident

Parameter	Value
Reactor Core Power, MWth	2754
Peaking Factor	1.7
Number of Failed Fuel Rods	176
Fuel Decay Time, hr	72
Fraction of Core Inventory in Fuel Gap Kr-85 I-131 Other Noble Gases Other Iodines	 0.20 0.16 0.10 0.10
Minimum Water Depth Above Damaged Fuel, ft	20.4
Iodine Decontamination Factor	120
Iodine Speciation Upon Release from Fuel Gap, % Elemental Organic	 97 3
Iodine Speciation Upon Release from Pool, % Elemental Organic	 82 18
Fuel Activity Release Duration, hr	2
Atmospheric Dispersion Factors	Tables 3.1.2 and 3.1.3

Table 3.2.3

Key Parameters Used in Radiological Consequence Analysis of Main Steam Line Break Accident

Parameter	Value
Reactor Core Power, MWth	2754
Peaking Factor	1.7
Failed Fuel, %	0.80
Primary RCS Equilibrium Activity, μCi/gm DE I-131 Preaccident Iodine Spike Factor Concurrent Iodine Release Rate Spike Factor	0.5 60 500
Secondary RCS Equilibrium Activity, μCi/gm DE I-131	0.1
Fraction of Core Inventory in Fuel Gap Kr-85 I-131 Other Noble Gases Other Iodines Alkali Metals	 0.20 0.16 0.10 0.10 0.24
Iodine Speciation from Failed Fuel, % Elemental Organic Aerosol/Particulate	 4.85 0.15 95.0
Iodine Speciation from Steam Generator, % Elemental Organic	 97 3
Time Until Shutdown Cooling is Established, hr	9
Steam Generator Coolant Volume, ft^3	4420.04
RCS Primary-to-Secondary Leak Rate, gpd	200
Steam Generator Release Rate, cfm	4000
Steam Generator Release Flashing, %	100
Atmospheric Dispersion Factors	Tables 3.1.2 and 3.1.3

Table 3.2.4

Key Parameters Used in Radiological Consequence Analysis of Steam Generator Tube Rupture Accident

Parameter	Value
Reactor Core Power, MWth	2754
Peaking Factor	1.7
Primary RCS Equilibrium Activity, µCi/gm DE I-131 Preaccident Iodine Spike Factor Concurrent Iodine Release Rate Spike Factor	0.5 60 335
Secondary RCS Equilibrium Activity, µCi/gm DE I-131	0.1
Iodine Speciation, % Elemental Organic	 97 3
Time Until Shutdown Cooling is Established, hr	8
RCS Primary-to-Secondary Leak Rate, gpd	200
Minimum Post-Accident Coolant Mass, lbm RCS Intact SG Ruptured SG	 391,900 56,420 124,644
Break Release to Intact SG	Time dependant values. See Letter dated March 22, 2007 (ML070810110), Attachment 1.
Break Release to Ruptured SG	Time dependant values. See Letter dated March 22, 2007 (ML070810110), Attachment 1.
Steam Release from Intact SG	Time dependant values. See Letter dated March 22, 2007 (ML070810110), Attachment 1.
Steam Release from Ruptured SG	Time dependant values. See Letter dated March 22, 2007 (ML070810110), Attachment 1.
Atmospheric Dispersion Factors	Tables 3.1.2 and 3.1.3

Table 3.2.5

Key Parameters Used in Radiological Consequence Analysis of Locked Rotor Accident

Parameter	Value
Reactor Core Power, MWth	2754
Peaking Factor	1.7
Failed Fuel, %	5.0
Fraction of Core Inventory in Fuel Gap Kr-85 I-131 Other Noble Gases Other Iodines Alkali Metals	 0.20 0.16 0.10 0.10 0.24
Iodine Speciation from Failed Fuel, % Elemental Organic Aerosol/Particulate	 4.85 0.15 95.0
Primary RCS Equilibrium Activity, µCi/gm DE I-131	0.5
Secondary RCS Equilibrium Activity, µCi/gm DE I-131	0.1
Time Until Shutdown Cooling is Established, hr	8
RCS Primary-to-Secondary Leak Rate, gpd	200
Coolant Mass Release, lbm 0 – 1800 seconds 1800 seconds – 8 hours	 204,500 1,416,625.98
RCS Iodine Release Flashing, % 0 – 15 minutes 15 minutes – 8 hours	 10 1
Atmospheric Dispersion Factors	Tables 3.1.2 and 3.1.3

Table 3.2.6

Key Parameters Used in Radiological Consequence Analysis of Control Rod Ejection Accident

Parameter	Value
Reactor Core Power, MWth	2754
Peaking Factor	1.7
Failed Fuel, %	
Incipient Centerline Melt and Cladding Failure	8
Cladding Failure	2
Total	10
Fraction of Core Inventory in Fuel Gap	
Noble Gas	0.10
Iodine	0.10
Alkali Metals	0.24
Iodine Speciation from Failed Fuel, %	
Elemental	4.85
Organic	0.15
Aerosol/Particulate	95.0
Primary RCS Equilibrium Activity, µCi/gm DE I-131	0.5
Secondary RCS Equilibrium Activity, µCi/gm DE I-131	0.1
Time Until Shutdown Cooling is Established, hr	8
Coolant Mass Release, lbm	
0 – 1800 seconds	204,500
1800 seconds – 8 hours	1,384,052.96
RCS Iodine Release Flashing, %	
0 – 15 minutes	10
15 minutes – 8 hours	1
RCS Primary-to-Secondary Leak Rate, gpd	200
Atmospheric Dispersion Factors	Tables 3.1.2 and 3.1.3

Table 3.2.7

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

Parameter	Value
Control Room Volume, ft^3	289,194
Recirculation Flow Rate, cfm	10,000 ± 10%
Emergency Ventilation System Initiation Delay, min	20
Recirculation Filter Efficiency, %	
Elemental	90
Organic	90
Aerosol/Particulate	99
Unfiltered Inleakage, cfm	3500
Occupancy Factors	
0 – 24 hours	1.0
24 – 96 hours	0.6
96 – 720 hours	0.4
Breathing Rate, m^3/sec	3.5E-04
Atmospheric Dispersion Factors	Table 3.1.2