Calvert Cliffs Nuclear Power Plant
Constellation Generation Group, LLC

1650 Calvert Cliffs Parkway
Lusby, Maryland 20657

Constellation Energy

November 3, 2005

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
License Amendment Request: Revision to Accident Source Term and Associated Technical Specifications

REFERENCES:

(a) J.J. DiNunno et al., Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," U.S. Atomic Energy Commission (now USNRC), 1962

(b) NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000

(c) Letter from G. Vanderheyden (CCNPP) to Document Control Desk (NRC), dated November 23, 2004, Supplemental Response to NRC Generic Letter 2003-01, "Control Room Habitability"

Pursuant to 10 CFR 50.90, Calvert Cliffs Nuclear Power Plant, Inc. hereby requests an amendment to Renewed Operating License Nos. DPR-53 and DPR-69. The proposed amendment will revise the accident source term in the design basis radiological consequence analyses in accordance with 10 CFR 50.67. The 10 CFR 50.67 requires that a licensee who seeks to revise its current accident source term to apply for a license amendment under 10 CFR 50.90. The proposed accident source term revision replaces the current methodology that is based on TID-14844 (Reference a) with the alternate source term methodology described in Regulatory Guide 1.183 (Reference b). This license amendment request is for full implementation of the alternate source term as described in Reference (b), with the exception that Reference (a) will continue to be used as the radiation dose basis for equipment qualification and vital area access. Additionally, the proposed amendment would revise the Calvert Cliffs Technical Specifications that are impacted by the proposed revision to the design basis accidents' source term.

This submittal fulfills our commitment for completing and submitting the analysis needed to meet Generic Letter 2003-01 objectives (Reference c).

The significant hazards discussion and the technical basis for this proposed change are provided in Attachment (1). Mark-ups of the affected Technical Specification pages are provided in Attachment (2). The Technical Specification Bases and the Updated Final Safety Analysis Report descriptions will be revised as appropriate to support the proposed license amendment request.

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In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated State of Maryland Official.

We indicated in Reference (c) that we would begin detailed engineering for the modifications to meet Generic Letter 2003-01 objectives after submittal of this proposed amendment, with an estimated completion date of September 24, 2007. However, since the proposed modifications are dependent upon the acceptability of the alternate source term methodology and could be subject to change during Nuclear Regulatory Commission Staff review, we will delay the start of the detailed engineering effort until consensus is reached with the Nuclear Regulatory Commission Staff on the appropriate scope of modifications to be performed for Generic Letter 2003-01. Therefore, as noted in Reference (c), we are unable to estimate an implementation completion date at this time.

Based on the above, we request approval of the proposed change as soon as possible for a timely completion of all of our commitments regarding Generic Letter 2003-01. We also request a 60 day implementation period for the approved amendment, following completion of required plant modifications, to allow sufficient time to implement procedure changes and training associated with this change.

Should you have questions regarding this matter, please contact Mr. L. S. Larragoite at (410) 495-4922.

Very truly yours,

STATE OF MARYLAND :
: **TO WIT:**
COUNTY OF CALVERT :

I, Bruce S. Montgomery, being duly sworn, state that I am Manager CCNPP Engineering Services - Calvert Cliffs Nuclear Power Plant, Inc. (CCNPP), and that I am duly authorized to execute and file this License Amendment Request on behalf of CCNPP. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other CCNPP employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of Calvert, this 3 day of November, 2005.

WITNESS my Hand and Notarial Seal:

Notary Public

My Commission Expires:

Wendy L. Hunter
NOTARY PUBLIC
Calvert County, Maryland
My Commission Expires 01/01/06

11/03/05
Date

BSM/GT/bjd

Attachments: (1) Technical Basis and No Significant Hazards Consideration

Enclosures:
1. CA06449 MHA Radiological Consequences Design Basis Calculation Using AST
2. CA06450 FHA Radiological Consequences Design Basis Calculation Using AST
3. CA06452 MSLB Radiological Consequences Design Basis Calculation Using AST
4. CA06453 SGTR Radiological Consequences Design Basis Calculation Using AST
5. CA06451 SRE Radiological Consequences Design Basis Calculation Using AST
6. CA06454 CEAEA Radiological Consequences Design Basis Calculation Using AST
7. CA06604 WGI Radiological Consequences Design Basis Calculation Using AST
8. CA06608 WPI Radiological Consequences Design Basis Calculation Using AST
9. CA06012 Atmospheric Dispersions Coefficient (X/Q) Calculation
10. CA06358 Source Terms Calculation
11. CA06421 Gas Gap Isotopic Fraction Calculation
12. CA06422 Primary and Secondary Isotopic Calculations
13. Compact Disk Containing Input Data for DBA Calculations

(2) Marked-up Technical Specification Pages

cc: P. D. Milano, NRC

(Attachments 1 and 2, without Enclosures)
S. J. Collins, NRC
Resident Inspector, NRC
R. I. McLean, DNR

ATTACHMENT (1)

TECHNICAL BASIS AND

NO SIGNIFICANT HAZARDS CONSIDERATION

TABLE OF CONTENTS

1. DESCRIPTION

The proposed amendment would revise the accident source term in design basis radiological consequence analyses and the associated Technical Specifications (TSs) for Calvert Cliffs Nuclear Power Plant (CCNPP) Units 1 and 2. The revision to the accident source term is needed to comply with the Control Room habitability regulatory requirement without relying on interim compensatory measures that are currently credited. The proposed TS revisions either support alternate source term evaluations or eliminate requirements that are no longer needed as a result of the revised design basis accident (DBA) analyses required to implement the alternate source term. This submittal fulfills our commitment in References (1) and (2) for completing and submitting the radiological analysis needed to meet Generic Letter 2003-01, "Control Room Habitability" objectives.

This application is submitted, in part, pursuant to Nuclear Regulatory Commission (NRC) regulation 10 CFR 50.67 that states, "[a] licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall apply for a license amendment under § 50.90." Section 50.67 further states, "[t]he application shall contain an evaluation of the consequences of applicable design basis accidents [ ] previously analyzed in the safety analysis report." Additionally, 10 CFR 50.67, sets a new acceptance criteria for radiological consequences based on total effective dose equivalent (TEDE), replacing the traditional whole body and thyroid dose guidelines stated in and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19 and 10 CFR 100.11. For CCNPP Units 1 and 2, the applicable DBAs that were re-analyzed for this application are:

a) Maximum Hypothetical Accident (MHA),
b) Fuel Handling Accident (FHA),
c) Main Steam Line Break (MSLB),
d) Steam Generator Tube Rupture (SGTR),
e) Seized Rotor Event (SRE),
f) Control Element Assembly Ejection Accident (CEAEA),
g) Waste Gas Incident (WGI), and
h) Waste Processing Incident (WPI).

The proposed accident source term revision follows the guidance in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 3). The accident source term discussed in Regulatory Guide 1.183 is herein referred to as the Alternate Source Term (AST). Regulatory Guide 1.183 permits full or selective implementation of the AST characteristics. This license amendment request is for full implementation of the AST as described in Reference (3) with the exception that the current methodology of Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" (Reference 4) will continue to be used as the radiation dose basis for equipment qualification and vital area access. Full implementation is a modification of the facility design basis that addresses all characteristics of the AST, that is, composition and magnitude of the radioactive material, its chemical and physical form, and the timing of its release. Full implementation revises the plant licensing basis to specify the AST in place of the previous accident source term and establishes the TEDE dose as the new acceptance criteria. This applies not only to the analyses performed in the application (which may only include a subset of the plant analyses), but also to all future design basis analyses.

2. PROPOSED CHANGE

The proposed license amendment would revise the CCNPP licensing basis to fully implement the Regulatory Guide 1.183 AST. As indicated in Section 1 above, implementation of AST for CCNPP consists of reevaluation of the applicable DBAs (MHA, FHA, MSLB, SGTR, SRE, CEAEA, WGI, and WPI) using the AST and the 10 CFR 50.67 TEDE acceptance criteria. In addition, the proposed license amendment would revise the following TSs that are associated with and justified by the analyses performed to support the AST. Attachment (2) contains the marked-up Technical Specification pages.

1. *Table of Contents (page iii)*

   Delete reference to Section 3.7.10 to reflect the deletion of TS requirement for Emergency Core Cooling System (ECCS) Pump Room Exhaust Filtration System (PREFS).

2. *TS Section 1.1. "Definitions" (page 1.1-3)*

   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.

3. *TS Section 1.1, "Definitions" (page 1.1-3)*

   The definition of L_a is revised to reflect the change in the maximum allowable containment leakage rate L_a used in the in the Containment Leakage Rate Testing Program (TS 5.5.16). L_a is reduced from 0.20 percent of containment air weight per day at P_a to 0.16 percent of containment air volume per day at P_a. The change from percent of containment air weight to percent of containment air volume is made for consistency with what is assumed in the accident analysis.

4. *TS Section 3.4.15, "RCS Specific Activity" (pages 3.4.15-1, 3.4.15-3, and 3.4.15-4)*

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

5. *TS Section 3.7.10, "Emergency Core Cooling System (ECCS) Pump Room Exhaust Filtration System (PREFS)" (pages 3.7.10-1 and 3.7.10-2)*

   The proposed amendment deletes TS 3.7.10.

6. *TS Section 3.7.11, "Spent Fuel Pool Exhaust Ventilation System (SFPEVS)" (pages 3.7.11-1 and 3.7.11-2)*

   In Limiting Condition for Operation (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).

7. *TS Section 3.9.3, "Containment Penetrations" (page 3.9.3-1)*

   In accordance with TSTF-312 (Reference 6), a note is added under 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.

8. *TS Section 5.5.11, "Ventilation Filter Testing Program" (pages 5.5-18 through 5.5.20)*

   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).

9. *TS Section 5.5.16, "Containment Leakage Rate Testing Program" (page 5.5-24)*

   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 volume per day at P_a. (Same change as Item 3 above.)

3. BACKGROUND

The current CCNPP licensing basis utilizes a source term determined in accordance with TID-14844 (Reference 4) to calculate offsite and Control Room doses for postulated design basis accidents. The interim CCNPP design basis radiological analysis that is currently credited for demonstrating Control Room habitability assumes that self-contained breathing apparatus (SCBA) is used as an interim compensatory measure. In response to NRC Generic Letter 2003-01 (References 1 and 2), we indicated that reanalysis of applicable accident scenarios in Chapter 14 of the CCNPP Updated Final Safety Analysis Report (UFSAR), using the AST per Regulatory Guide 1.183, combined with some plant modifications will be needed to retire the interim compensatory measure. To that end, this submittal contains all the required reanalysis and the licensing basis changes to retire these compensatory measures.

The fission product release from the reactor core into Containment is referred to as the "source term," and it is characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release from the reactor core. Since the publication of Reference (4), significant advances have been made in understanding the composition and magnitude, chemical form, and timing of fission product releases from severe nuclear power plant accidents. Many of these insights developed out of the major research efforts started by the NRC and the nuclear industry after the accident at Three Mile Island.

In 1995, NUREG-1465 (Reference 5) was published with revised ASTs for use in the licensing of future Light Water Reactors (LWRs). This NUREG represents the result of decades of research on fission product release and transport in LWRs under accident conditions. On December 23, 1999, the NRC issued the final rule on "Use of Alternate Source Terms at Operating Reactors." The final rule, issued as 10 CFR 50.67, "Accident Source Term," allows holders of operating licenses issued prior to January 10, 1997, to voluntarily replace the traditional source term used in DBA analyses with alternative source terms such as the one described in NUREG-1465. One of the major insights summarized in NUREG-1465 involves the timing and duration of fission product releases.

The five release phases describing the progression of a severe accident in a LWR are listed in NUREG-1465 and are given below.

1. Coolant Activity Release
2. Gap Activity Release
3. Early In-vessel Release
4. Ex-vessel Release
5. Late In-vessel Release

Phases 1, 2, and 3 are considered in current (i.e., pre-AST) DBA evaluations; however, they are all assumed to occur instantaneously. Phases 4 and 5 are related to severe accident evaluations. Under the AST methodology, only the coolant activity release (i.e., Phase 1) is assumed to occur instantaneously and end with the onset of the gap activity release (i.e., Phase 2). This approach represents a more realistic time sequence for activity release. The insights from NUREG-1465 were subsequently incorporated into Regulatory Guide 1.183 (Reference 3).

4. TECHNICAL ANALYSIS

4.1 Design Basis Accidents Re-analysis Using AST

As justification for the proposed revision to CCNPP licensing basis to implement the AST, this section provides the results and brief summaries of the eight radiological consequence DBAs (MHA, FHA, MSLB, SGTR, SRE, CEAEA, WGI, WPI) that were re-analyzed. The detailed calculations that contain data input, assumptions and analyses methodologies are provided in Enclosures (1) through (8) of this attachment.

These analyses have incorporated the Regulatory Guide 1.183 features of the AST, including the TEDE analysis methodology and modeling of plant systems and equipment operation that influence the events. The calculated radiological consequences are compared with the revised limits provided in 10 CFR 50.67(b)(2), and as clarified per the additional guidance in Regulatory Guide 1.183 for events with a higher probability of occurrence.

Dose calculations are performed for the Exclusion Area Boundary (EAB) for the worst 2-hour period, and for the Low Population Zone (LPZ) and Control Room for the 30-day duration of the accident. All of the radiological dose analyses for the eight DBAs were performed utilizing the RADTRAD computer code. The RADTRAD computer code calculates the Control Room and offsite doses resulting from releases of radioactive isotopes based on user supplied atmospheric dispersion factors, breathing rates, occupancy factors, and dose conversion factors. The RADTRAD computer code models the transport of radioactivity (elemental, particulate, and noble gas radionuclides) from the sprayed and unsprayed regions of a primary containment, spent fuel pool (SFP), or other area where a release can occur, through the secondary containment if any, and then to the environment and to the Control Room. The code includes the capability to model time-dependent activity release; containment spray, filtration, and leakage; Control Room filtration and in-leakage; primary and secondary containment purge filters; Control Room intake and recirculation filters; atmospheric dispersion; and natural decay. References for qualification of the RADTRAD computer code and all other computer codes used are provided with each design basis calculation in Enclosures (1) through (8).

Enclosure (9) contains the CCNPP Control Room atmospheric dispersion factor (X/Q) calculation used in the re-analysis of all the radiological consequence DBAs. The atmospheric dispersion factors were calculated using the computer code ARCON96. ARCON96 implements a computational model for calculating atmospheric dispersion coefficients in the vicinity of buildings. The model estimates impacts from ground level, vent, and elevated releases using a single-year or multi-years of hourly meteorological data. This model also treats diffusion more realistically under low wind speed conditions than previous NRC-issued models. The meteorological data used in the calculation includes eight years (1991-1998) of hourly wind speed, wind direction, and stability class readings from the CCNPP meteorological tower.

The EAB and LPZ atmospheric dispersion factors used in the re-analysis of all the radiological consequences DBAs are obtained from the CCNPP UFSAR. These dispersion factors are part of the existing design basis offsite dose calculations.

Enclosure (10) contains the source term calculation for the isotopic activities released from the failed fuel. The isotopic activities were generated utilizing the isotope generation and depletion computer code SAS2H/ORIGEN-S which is part of the SCALE 4.4 code system. For those DBAs in which cladding failure releases the gas gap activity of the affected fuel rods, the gas gap fractions were calculated in Enclosure (11). Regulatory Guide 1.183 requires that if no or minimal fuel damage is postulated for a particular DBA, the activity released should be the maximum coolant activity allowed by the TSs or the maximum coolant activity generated by a preexisting or concurrent iodine spike. Accordingly, Enclosure (12) contains the source term primary and secondary isotopic calculations for a design basis event that assumes no or minimal fuel damage. Enclosure (13) provides a compact disk containing the input data for all calculations performed to support the DBA reanalysis.

4.1.1 Maximum Hypothetical Accident (MHA)

Section 14.24 of the CCNPP UFSAR describes the MHA. The maximum hypothetical accident is a non-mechanistic scenario which evaluates the Containments' capability to contain released radioisotopes. Safety system effectiveness is not considered; the quantity of radioisotopes released to the containment atmosphere is dependent on the power level (MWt) of the reactor. The criteria for this release are established so that the magnitude of the release bounds all credible accident releases.

Enclosure (1) contains the detailed MHA radiological consequences design basis calculation using the AST. The following supporting calculations are also provided: Atmospheric Dispersion Coefficient (X/Q) calculation (Enclosure 9), Source Terms calculation (Enclosure 10), and Primary and Secondary Isotopic calculations (Enclosure 12).

The MHA re-analysis for AST implementation considered the following radiological release pathways for offsite and Control Room doses.

- Containment pathway
- Hydrogen Purge line pathway
- Ventilation stack pathway
- Refueling Water Tank pathway
- Containment shine
- Plume shine
- Control Room Filter shine

Major assumptions and required plant modifications considered in the MHA re-analysis to meet regulatory requirements are:

- A bounding Control Room in-leakage value of 3,500 cfm,
- Modification of the CREVS to a nominal 10,000 cfm flow with 90 percent filtration efficiency for elemental and organic iodine and 99 percent for particulate iodine was credited,
- Installation of automatic isolation dampers and radiation monitors at Access Control HVAC Unit RTU-1 and Access Control Air Conditioning Unit 13 on the Auxiliary Building roof,
- Revision to the TS 3.4.15 limit for RCS activity from 1.0 μCi/gm to 0.5 μCi/gm,
- Revision to the TS 5.5.16 maximum allowable containment leakage rate, L_a, from 0.20 percent of containment air weight per day at P_a to 0.16 percent of containment air volume per day at P_a,
- Sump pH neutrality and iodine re-evolution from the sump is controlled by neutral pH which is maintained by adequate trisodium phosphate dodecahydrate (TSP) in the Containment. The mass of TSP required to neutralize the containment sump pH following a loss-of-coolant accident (LOCA) is calculated assuming a boron level of 3105.5 ppm. This mass of TSP is converted into a TSP volume of 289.3 ft^3 (TS 3.5.5 LCO limit for TSP volume). Note that the dodecahydrate

form of TSP is used because of the high humidity in the Containment during normal operation. Since the TSP is hydrated, it is less likely to absorb large amounts of water from the humid atmosphere and will undergo less physical and chemical change than the anhydrous form of TSP, and

- Additional sources of acid in the containment sump post-LOCA were identified and analyzed per NUREG/CR-5950 "Iodine Evolution and pH Control" including nitric acid generated by irradiation of air and water in Containment, hydrochloric acid generated by the breakdown of certain cable jacketing when it is exposed to high temperatures and radiation, hydriodic acid generated by a reaction of iodine in water, and carbonic acid generated by water absorbing CO_2 from the air or from limestone concrete. The increase in TSP required to neutralize the additional acid sources was found to be insignificant and to have a negligible effect on pH.

Table 1 summarizes the results of the MHA re-analysis for the EAB, LPZ, and Control Room doses. The corresponding regulatory dose limits are shown in the last row of the table. All calculated doses are less than the regulatory limit.

Table 1

MHA Results

Cases	EAB (Rem)	LPZ (Rem)	Control Room (Rem)
Containment Pathway	1.8988	0.4958	3.9682
Penetration Room Pathway	0.1838	0.0485	0.3968
Refueling Water Tank Pathway	1.9676E-05	1.8560E-03	3.2979E-01
Hydrogen Purge Pathway	6.4918E-05	1.5283E-05	7.7048E-05
Containment Shine			0.0547
Plume Shine			0.0030
Control Room Filter Shine			0.0139
Total	**2.0827**	**0.5462**	**4.7664**
Regulatory Limits	**25**	**25**	**5**

4.1.2 Fuel Handling Accident (FHA)

Section 14.18 of the CCNPP UFSAR describes the design basis FHA, which is assumed to occur in the SFP and the refueling pool (RFP). The analyses for a FHA in the SFP and the RFP both assume that gas gap activity from 176 fuel rods of the highest power assembly is released immediately and uniformly throughout the area. The FHA analysis assumes a total iodine decontamination factor (DF) of 200 based on a minimum water depth of 23'. In the RFP this assumption is preserved by the TS requirement of 23' of water above fuel assemblies seated in the reactor core. In the SFP, the TS requires 21.5' of water above fuel assemblies seated in the SFP storage racks. This TS was deemed sufficient to preserve the required 23' of water because a FHA was assumed to occur as a fuel assembly strikes the bottom of the SFP.

However, the most limiting condition for SFP is spent fuel inspection and reconstitution where assemblies are placed on spacers. When assemblies are placed on rack spacers and their upper end fittings are removed, a FHA from a dropped heavy object would require a lower DF based on reduced water coverage. A revised DF of 120 for a FHA during reconstitution/inspection with 20.4' of water between the top of the pin and the surface of the water was computed for a 20.5" rack spacer.

The air in both the RFP and SFP areas is monitored. The SFP ventilation system draws air across the SFP area; this air is discharged unfiltered to the atmosphere through the plant vent. The SFP ventilation system processes 32,000 ± 10% cfm of the SFP volume into the environment with no credit for the high efficiency particulate air filters or charcoal filters for the duration of the accident. In this case the activity escapes to the environment assuming complete release in two hours and is transported to the EAB and to the Control Room via appropriate atmospheric dispersion coefficients. After a FHA in Containment, the activity may be released through the personnel air lock, the containment outage door, the containment walls themselves, or via the hydrogen or 48" purge lines into the plant vent. The release through the plant vent is most limiting, therefore, the activity from a FHA in the Containment or the SFP will be assumed to be released unfiltered to the environment through the plant vent stack. Note also that this analysis bounds penetration flow paths that have direct access from the containment atmosphere to the outside atmosphere which could be un-isolated under administrative control. Since this current analysis assumes that the radioactive release is unfiltered, completely released over a two hour time period, and released with the most limiting dispersion coefficients, the analysis also applies to the containment penetration flow paths that are opened under administrative control.

Enclosure (2) contains the detailed FHA radiological consequences design basis calculation using the AST. The following supporting calculations are also provided: Atmospheric Dispersion Coefficient (X/Q) calculation (Enclosure 9), Source Terms calculation (Enclosure 10), and Gas Gap Isotopic Fraction calculations (Enclosure 11).

Major assumptions and required plant modifications considered in the FHA re-analysis include:

- A bounding Control Room in-leakage value of 3,500 cfm,
- Modification of the CREVS to a nominal 10,000 cfm flow with 90 percent filtration efficiency for elemental and organic iodine and 99 percent for particulate iodine was credited,
- Installation of automatic isolation dampers and radiation monitors at Access Control HVAC Unit RTU-1 and Access Control Air Conditioning Unit 13 on the Auxiliary Building roof,
- Fuel movement does not occur until 72 hours after reactor shutdown. Under current requirements (Technical Requirements Manual 15.9.1), fuel movement can occur 100 hours after reactor shutdown. This requirement may be decreased to 72 hours following approval of this analysis.

Table 2 summarizes the results of the FHA re-analysis for the EAB, LPZ, and Control Room doses. The corresponding regulatory dose limits are shown in the last row of the table. All calculated doses are less than the regulatory limit.

Table 2

FHA Results

Cases	EAB (Rem)	LPZ (Rem)	Control Room (Rem)
Containment/SFP[1]	0.6958	0.1638	2.3314
SFP[2]	1.1136	0.2622	3.8538
Regulatory Limits	**6.3 (RG 1.183)**	**6.3 (RG 1.183)**	**5 (10 CFR 50.67)**

(1) Assumes a decay time of 72 hours and a DF of 200.
(2) Assumes a decay time of 72 hours and a DF of 120.

4.1.3 Main Steam Line Break (MSLB)

Section 14.14 of the CCNPP UFSAR describes the design basis MSLB Event. For this analysis, a MSLB Event is defined as the pre-trip guillotine-type rupture of a main steam line outside Containment in the main steam piping room. It is assumed that the steam line rupture occurs between the steam generator (SG) and the main steam isolation valve, allowing blowdown of the affected SG to continue. A loss of offsite power (LOOP) with the turbine trip results in the maximum site boundary doses, since the LOOP causes the reactor coolant pumps (RCPs) to coast down, minimizing core flow, lowering the transient Departure from Nucleate Boiling Ratio, and maximizing the number of fuel pins predicted to fail.

To maximize Control Room and offsite doses, the maximum secondary TS activity and that fraction of the primary TS and failed fuel activity which leaks to the secondary from the affected and unaffected SGs are discharged into the main steam piping room and out the main steam piping room vent (gooseneck) on the roof of the Auxiliary Building. Since the SGs are designed to withstand RCS operating pressure on the tube side with atmospheric pressure on the shell side, the continued integrity of the RCS barrier is assured. Thus, only the maximum TS primary-to-secondary leakage is assumed. Per Regulatory Guide 1.183, a partition factor of unity is assumed for all discharged activities.

Enclosure (3) contains the detailed MSLB radiological consequences design basis calculation using AST. The following supporting calculations are also provided: Atmospheric Dispersion Coefficient (X/Q) calculation (Enclosure 9), Source Terms calculation (Enclosure 10), Gas Gap Isotopic Fraction calculation (Enclosure 11), and Primary and Secondary Isotopic calculations (Enclosure 12).

Major assumptions and required plant modifications considered in the MSLB re-analysis include:

- A bounding Control Room in-leakage value of 3,500 cfm,
- Modification of the CREVS to a nominal 10,000 cfm flow with 90 percent filtration efficiency for elemental and organic iodine and 99 percent for particulate iodine was credited,
- Installation of automatic isolation dampers and radiation monitors at Access Control HVAC Unit RTU-1 and Access Control Air Conditioning Unit 13 on the Auxiliary Building roof,
- Revision to the TS 3.4.15 limit for RCS activity from 1.0 μCi/gm to 0.5 μCi/gm

Per Regulatory Guide 1.183 (Reference 3), if no or minimal fuel damage is postulated, the activity should be the maximum coolant activity allowed by the TSs, assuming two cases of iodine spiking. Thus, three cases are modeled to determine the maximum offsite and Control Room doses:

- Failed Fuel Case: Assuming 0.80 percent of the fuel pins in the core failed (Note that the MSLB analyses of record generated by Westinghouse results in no predicted failed fuel.)
- Preaccident Iodine Spike (PIS) Case: A reactor transient is assumed to occur prior to the postulated MSLB and has raised the primary coolant iodine concentration to the maximum value permitted by the TS, which is 60 times the new TS 3.4.15 limit of 0.5 μCi/gm.
- Concurrent Iodine Spike (CIS) Case: The primary system transient associated with the MSLB is assumed to cause an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes the iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value.

Table 3 summarizes the results of the MSLB re-analysis for the EAB, LPZ, and Control Room doses. Table 3 provides the failed fuel case and the PIS and CIS cases. The corresponding regulatory dose limits are also shown. The calculated doses for all three cases are less than the regulatory limits.

Table 3

MSLB Results

Cases	EAB (Rem)	LPZ (Rem)	Control Room (Rem)
0.8% Failed Fuel	0.2180	0.0577	4.6301
PIS Case	3.2635E-03	9.2117E-04	8.2508E-02
Regulatory Limits	**25 (RG 1.183)**	**25 (RG 1.183)**	**5 (10 CFR 50.67)**
CIS Case	2.2499E-03	1.0446E-03	2.0757E-01
Regulatory Limits	**2.5 (RG 1.183)**	**2.5 (RG 1.183)**	**5 (10 CFR 50.67)**

4.1.4 Steam Generator Tube Rupture (SGTR)

Section 14.15 of the CCNPP UFSAR describes the design basis evaluation of the SGTR Event. A SGTR Event is defined as the penetration of the barrier between the RCS and the main steam system. The integrity of this barrier is of radiological safety significance, in that a leaking SG tube allows the transfer of reactor coolant into the main steam system. Radioactivity contained in the reactor coolant would then mix with the fluid in the secondary side of the affected SG. This radioactivity would then be transported by steam to the turbine/condenser/vent stack/atmosphere or directly to the atmosphere via the atmospheric dump valves (ADVs) or main steam safety valves.

The limiting SGTR Event is considered to be a complete double-ended tube break and 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 Event allows primary coolant to leak into the secondary side via the SG. The current design basis accident assumes that cooldown of the RCS to shutdown cooling (SDC) conditions utilizes the ADVs of both SGs until SDC conditions are attained. In this analysis, the ADV of the affected SG must be isolated after two hours to limit activity emissions. The following three cooldown operational modes are considered:

- The operator continues the cooldown via the ADV of the unaffected SG until the 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. Note that a 30 day cooldown via the ADV of the unaffected SG is conservatively modeled in this analysis.
- The operator continues the cooldown via the ADV of the unaffected SG but also establishes conditions that allow the use of the SG blowdown system during the cooldown phase of the event. Note that use of SG blowdown can be assumed to occur at any time during the cooldown phase of the event. However, the offsite and Control Room doses are bounded by those of the first option.
- 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.

Enclosure (4) contains the detailed SGTR radiological consequences design basis calculation using the AST. The following supporting calculations are also provided: Atmospheric Dispersion Coefficient (X/Q) calculation (Enclosure 9), Source Terms calculation (Enclosure 10), and Primary and Secondary Isotopic calculations (Enclosure 12).

Major assumptions and required plant modifications considered in the SGTR re-analysis include:

- A bounding Control Room in-leakage value of 3,500 cfm,
- Modification of the CREVS to a nominal 10,000 cfm flow with 90 percent filtration efficiency for elemental and organic iodine and 99 percent for particulate iodine was credited,
- Installation of automatic isolation dampers and radiation monitors at Access Control HVAC Unit RTU-1 and Access Control Air Conditioning Unit 13 on the Auxiliary Building roof,
- Revision to the TS 3.4.15 limit for RCS activity from 1.0 μCi/gm to 0.5 μCi/gm.

Per Regulatory Guide 1.183 (Reference 3), if no or minimal fuel damage is postulated, the activity should be the maximum coolant activity allowed by the TSs, assuming two cases of iodine spiking. Thus, two cases are modeled to determine the maximum offsite and Control Room doses.

- PIS Case: A reactor transient is assumed to occur prior to the postulated SGTR and has raised the primary coolant iodine concentration to the maximum value permitted by the TSs, which is 60 times the new TS 3.4.15 limit of 0.5 μCi/gm.
- CIS Case: The primary system transient associated with the SGTR is assumed to cause an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes the iodine release rate from the fuel rods to the primary coolant increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value. Per Regulatory Guide 1.183, the assumed iodine spike duration should be 8 hours.

Table 4 summarizes the results of the SGTR re-analysis for the EAB, LPZ, and Control Room doses for the design-basis CIS and PIS cases for the three operational cooldown modes described above [cooldown via ADV of unaffected SG from 2 hours to 30 days, additional cooldown via the ADV of the affected SG from 0-2 and 24-32 hours, and cooldown via the unaffected SG with blowdown to the Waste Processing System (WPS)]. The corresponding regulatory dose limits are also shown. The calculated doses in all cases are less than the regulatory limit.

Table 4

SGTR Results

Cases	EAB (Rem)	LPZ (Rem)	Control Room (Rem)
CIS Unaffected ADV 0-2 hr	0.1964	0.0484	1.7081
CIS Affected ADV 0-2/24-32 hr	0.1964	0.0476	1.6929
CIS Unaffected ADV 0-2 hr/WPS	0.1964	0.0484	1.7081
CIS Regulatory Limits	**2.5**	**2.5**	**5**
PIS Unaffected ADV 0-2 hr	0.4910	0.1164	4.1590
PIS Affected ADV 0-2/24-32 hr	0.4910	0.1162	4.1655
PIS Unaffected ADV 0-2 hr/WPS	0.4910	0.1164	4.1590
PIS Regulatory Limits	**25**	**25**	**5**

4.1.5 Seized Rotor Event (SRE)

Section 14.16 of the CCNPP UFSAR describes the design basis evaluation of the SRE. A SRE is defined as a complete seizure of a single RCP shaft. The seizure is postulated to occur due to a mechanical failure or a loss of component cooling to the pump shaft seals. The most limiting SRE is an instantaneous RCP shaft seizure at hot full power. 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. 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, the core power will increase. The core average heat flux will decrease slightly due to the increasing core temperatures. The insertion of the control element assemblies (CEAs) due to a low RCS flow trip will terminate the power rise; however, a limited number of fuel pins will experience departure from nucleate boiling for a short period of time and are thus predicted to fail. The initial secondary activity together with initial primary activity and failed fuel activity released to the primary system that then leaks into the secondary system will escape out of the SGs via the atmospheric dump valves. Note that per the requirements of Regulatory Guide 1.183 (Reference 3), the release of fission products from the secondary system should be evaluated with the assumption of a coincident LOOP. Thus, the use of condensers can not be credited in this analysis.

Enclosure (5) contains the detailed SRE radiological consequences design basis calculation using the AST. The following supporting calculations are also provided: Atmospheric Dispersion Coefficient (X/Q) calculation (Enclosure 9), Source Terms calculation (Enclosure 10), Gas Gap Isotopic Fraction calculation (Enclosure 11), and Primary and Secondary Isotopic calculations (Enclosure 12).

Major assumptions and required plant modifications considered in the SRE re-analysis include:

- A bounding Control Room in-leakage value of 3,500 cfm,
- Modification of the CREVS to a nominal 10,000 cfm flow with 90 percent filtration efficiency for elemental and organic iodine and 99 percent for particulate iodine was credited,
- Installation of automatic isolation dampers and radiation monitors at Access Control HVAC Unit RTU-1 and Access Control Air Conditioning Unit 13 on the Auxiliary Building roof,
- Revision to the TS 3.4.15 limit for RCS activity from 1.0 μCi/gm to 0.5 μCi/gm.

Two SRE models were constructed: a two-hour SG release model to maximize the EAB dose and an eight-hour SG release model to maximize the Control Room dose. Each SRE model is composed of six release components: gas gap iodine releases, gas gap noble gas releases, gas gap alkali metal releases, TS primary iodine activity releases, TS primary noble gas activity releases, and TS secondary iodine activity releases. The SRE is assumed to occur at time t=0 releasing the failed fuel gas gap iodine, noble gas, and alkali metal activities immediately and homogeneously into the primary system.

Table 5 summarizes the results of the SRE re-analysis for the EAB, LPZ, and Control Room doses. The corresponding regulatory dose limits are also shown in the last row of the table. The calculated doses for both the eight-hour and two-hour cases are less than the regulatory limit.

Table 5

SRE Results

Cases	EAB (Rem)	LPZ (Rem)	Control Room (Rem)
8 Hour Secondary Pathway	0.0336	0.0095	0.7885
2 Hour Secondary Pathway	0.0386	0.0091	0.2294
Regulatory Limits	**2.5 (RG 1.183)**	**2.5 (RG 1.183)**	**5 (10 CFR 50.67)**

4.1.6 Control Element Assembly Ejection Accident (CEAEA)

Section 14.13 of the CCNPP UFSAR describes the design basis evaluation of the CEAEA. A CEAEA is defined 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. The event is postulated to occur as a result of a complete, instantaneous, circumferential rupture of either the control element drive mechanism pressure housing or the control element drive mechanism nozzle from the reactor vessel closure head. The pressure of the RCS causes the ejection of the extension shaft through the rupture and the movement of the CEA to a fully-withdrawn position. The most limiting CEAEA is a 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 an exponential increase in nuclear power.

Enclosure (6) contains the detailed CEAEA radiological consequences design basis calculation using the AST. The following supporting calculations are also provided: Atmospheric Dispersion Coefficient (X/Q) calculation (Enclosure 9), Source Terms calculation (Enclosure 10), and Primary and Secondary Isotopic calculations (Enclosure 12).

Major assumptions and required plant modifications considered in the CEAEA re-analysis to meet regulatory requirements are:

- A bounding Control Room in-leakage value of 3,500 cfm,
- Modification of the CREVS to a nominal 10,000 cfm flow with 90 percent filtration efficiency for elemental and organic iodine and 99 percent for particulate iodine was credited,
- Installation of automatic isolation dampers and radiation monitors at Access Control HVAC Unit RTU-1 and Access Control Air Conditioning Unit 13 on the Auxiliary Building roof,
- Revision to the TS 3.4.15 limit for RCS activity from 1.0 μCi/gm to 0.5 μCi/gm,
- Revision to the TS 5.5.16 maximum allowable containment leakage rate, L_a, from 0.20 percent of containment air weight per day at P_a to 0.16 percent of containment air volume per day at P_a.

The CEAEA was analyzed for a 30-day containment pathway, a 2-hour to shutdown cooling secondary pathway, and an 8-hour to shutdown cooling secondary pathway. Table 6 summarizes the results of the CEAEA re-analysis for the EAB, LPZ, and Control Room doses. The corresponding regulatory dose limits are shown in the last row of the table. All calculated doses are less than the regulatory limit.

Table 6

CEAEA Results

Results	EAB (Rem)	LPZ (Rem)	Control Room (Rem)
8 hour Secondary Pathway	0.32616	0.08735	4.59461
2 hour Secondary Pathway	0.35658	0.08394	1.45292
30 day Containment Pathway	0.4567	0.1187	0.9679
Regulatory Limits	**6.3**	**6.3**	**5**

4.1.7 Waste Gas Incident (WGI)

Section 14.22 the CCNPP UFSAR describes the design basis evaluation of the WGI. The most limiting WGI is defined as an unexpected and uncontrolled release to the atmosphere of the radioactive xenon and krypton fission gases that are stored in one waste gas decay tank. As the components of the waste gas system are subjected to pressures no greater than 150 psig, a failure is not likely. However, a non-mechanistic rupture of a waste gas decay tank is analyzed to define the limit of the hazard that could result from any malfunction in the radioactive waste gas system.

Enclosure (7) contains the detailed WGI radiological consequences design basis calculation using the AST. The following supporting calculations are also provided: Atmospheric Dispersion Coefficient (X/Q) calculation (Enclosure 9), Source Terms calculation (Enclosure 10), and Primary and Secondary Isotopic calculations (Enclosure 12).

Major assumptions and required plant modifications considered in the WGI re-analysis include:

- A bounding Control Room in-leakage value of 3,500 cfm,
- Installation of automatic isolation dampers and radiation monitors at Access Control HVAC Unit RTU-1 and Access Control Air Conditioning Unit 13 on the Auxiliary Building roof,
- The maximum activity in one waste gas decay tank occurs from shutdown and degassing of the primary system for one unit that has been operating for an extended period of time with primary noble gas activity at the TS 3.4.15 limit of 100/Ebar μCi/gm, which bounds the equilibrium noble gas activity for 1% failed fuel.

The RADTRAD calculated TEDE dose for each location assuming 4 hours credit for radioactive decay during the 9 to 11 hour degassing process is presented in Table 7 below. Regulatory Guide 1.183 does not provide TEDE dose acceptance criteria for the WGI. The limits in 10 CFR 20.1301(a)(1) will be utilized for the EAB and LPZ TEDE dose limits for this event. The WGI results in Table 7 are below the revised EAB TEDE limit. The 0-30 day Control Room dose is below the 5 rem TEDE dose limit established in 10 CFR 50.67(b)(2)(iii).

Based on meeting the above mentioned dose limits, this evaluation affirms that all regulatory limits are met.

Table 7

WGI Results

Case	EAB (rem)	LPZ (rem)	Control Room (rem)
4-Hour Gas Transfer Credit	0.098	0.023	0.062
Regulatory Limit	**0.1**	**0.1**	**5**

4.1.8 Waste Processing Incident (WPI)

Section 14.23 of the CCNPP UFSAR describes the design basis evaluation of the WPI used to determine the seismic design classification of waste processing components per NRC Safety Guide 29, "Seismic Design Classification." The WPI is a seismically-induced failure of the reactor coolant WPS located in the Auxiliary Building. It is hypothesized that seismic induced failure of a WPS component would release the contents into the Auxiliary Building, where curbs and floor drains would direct the fluid from the failed components to the miscellaneous waste receiver tank (-5' elevation) and any airborne component would be drawn into the ventilation system and discharged to the atmosphere through the plant vent stack. The activity of the fluid entering the WPS is based on reactor coolant with an activity bounding both 1% failed fuel and TS limits. Decontamination factors consistent with NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors" are utilized in determining the activity in WPS components. A noble gas DF of 6 (which is less than the measured DF) is conservatively utilized for the reactor coolant bleed degasifiers. No credit is taken for radioactive decay during the time required for the processed fluid to reach the component, with the exception of one of two receiver tanks and both monitor tanks, for which 300 hours of decay is taken. No decay time is conservatively credited for the receiver tank being actively filled or other structures, systems, and components in the WPS.

Enclosure (8) contains the detailed WPI radiological consequences design basis calculation using the AST. The following supporting calculations are also provided: Atmospheric Dispersion Coefficient (X/Q) calculation (Enclosure 9), Source Terms calculation (Enclosure 10), and Primary and Secondary Isotopic calculations (Enclosure 12).

Major assumptions and required plant modifications considered in the WPI re-analysis include:

- A bounding Control Room in-leakage value of 3,500 cfm,
- The reactor coolant waste evaporators will be retired-in-place,
- Installation of automatic isolation dampers and radiation monitors at Access Control HVAC Unit RTU-1 and Access Control Air Conditioning Unit 13 on the Auxiliary Building roof,
- No flashing of the fluid in the WPS occurs upon failure, and release fractions for tritium, iodine, and noble gases are assumed to be 10%, 10.1%, and 100%, respectively. The release is conservatively treated as an instantaneous puff release. In addition, 10% of the equilibrium halogen inventory on the resin of failed ion exchangers is assumed to instantaneously and non-mechanistically transfer to the water,
- The WPS is designed to process 28 RCS volumes (14 per unit) per year (UFSAR Section 11.1.2.1.1).

The RADTRAD-calculated TEDE dose for the WPI is presented in Table 8 below. Regulatory Guide 1.183 does not provide TEDE dose acceptance criteria for the WPI. The limit in 10 CFR 20.1301(a)(1) will be utilized for the EAB and LPZ TEDE dose limits for this event. The WPI results in Table 8 are below the revised EAB TEDE limit. The 0-30 day Control Room dose is below the 5 rem TEDE dose limit established in 10 CFR 50.67(b)(2)(iii).

Based on meeting the above mentioned dose limits, Seismic Category I designation is not required for the WPS components considered. However, the current seismic classification will remain unchanged from that described in UFSAR Section 5A.2.1.2.

Table 8
WPI Results

Cases	EAB (rem)	LPZ (rem)	Control Room (rem)
Degasifier Noble Gas DF = 6; 300h decay for three-of-four 90k gal tanks	0.092	0.022	0.314
Regulatory Limit	**0.1**	**0.1**	**5**

4.1.9 Conclusion

The revised DBA analyses described herein have incorporated the Regulatory Guide 1.183 features of the AST, including the TEDE analysis methodology and modeling of plant systems and equipment operation that influence the events. The calculated radiological consequences for the EAB, LPZ, and Control Room are compared with the revised limits provided in 10 CFR 50.67(b)(2), and as clarified per the additional guidance in Regulatory Guide 1.183 for events with a higher probability of occurrence. The estimated dose consequences for all design basis DBAs meet the acceptance criteria specified in 10 CFR 50.67 (as clarified by Regulatory Guide 1.183), or 10 CFR 20.1301(a)(1), as appropriate. This represents a full implementation of the AST in which the Regulatory Guide 1.183 accident source term will become the licensing basis for CCNPP with the exception that TID-14844 will continue to be used as the radiation dose basis for equipment qualification and vital area access.

4.2 Proposed Revision to the Technical Specifications

This section provides the justification for the proposed revisions to the TSs that are associated with the proposed licensing basis revision to implement the AST.

4.2.1 Revision to the Definition of DOSE EQUIVALENT I-131

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 instead of the current TID-14844 inhalation dose conversion factors. Per Regulatory Guide 1.183, Sections 4.1.1 and 4.1.2, the AST implementation analyses described in Section 4.1 above use the thyroid conversion factors listed in Table 2.1 of Federal Guidance Report 11. Thus, this proposed revision is supported by the justification for the proposed licensing basis revision to implement the AST.

4.2.2 Revision to the Maximum Allowable Containment Leakage Rate, La

The maximum allowable containment leakage rate L_a contained in TS 1.1, "Definitions" and 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 volume per day at P_a. This revision is required to meet the Control Room regulatory dose limit for the design basis MHA described in Section 4.1.1 above. The revised leakage rate is also used in the CEA Ejection Event analysis; however, the lower leakage limit is not necessary to meet regulatory requirements for that accident. From the TS requirement and safety perspective, this change is more conservative than the existing requirement. Examination of CCNPP Unit 1 and Unit 2 Integrated Leak Rate Test data indicate that except for single tests for Unit 1 in 1978 and for Unit 2 in 1979, all values of L_a remain well below the revised acceptance limits. All containment leakage rate tests since 1980 are below 0.12 percent. Therefore, achieving the proposed more restrictive leakage rate will not create additional hardship.

4.2.3 Revision to the RCS Specific Activity

The limit for RCS activity in TS 3.4.15 "RCS Specific Activity" was reduced from 1.0 μCi/gm DOSE EQUIVALENT I-131 to 0.5 μCi/gm DOSE EQUIVALENT I-131. This revision is required to meet Control Room dose regulatory limits for the design basis SGTR accident described in Section 4.1.4 above. This revised RCS specific activity limit is also used in the maximum hypothetical accident, CEA ejection event, MSLB, and SRE accident analyses; however, the lower activity limit is not necessary to meet regulatory requirements for these accidents. From the TS requirement and safety perspective, this change is more conservative than the existing requirement. Examination of CCNPP Unit 2 Cycles 13 and 14 and Unit 1 Cycles 15 and 16 DOSE EQUIVALENT I-131 (DEQ I-131) indicates that the DEQ I-131 remains well below the proposed revised limits. For Unit 1 and Unit 2, the equilibrium values are below 0.02 μCi/gm. Therefore, achieving the proposed more restrictive RCS activity will not create additional hardship.

4.2.4 Deletion of the TS Requirements for PREFS

The proposed revision deletes the LCO, Actions, and the associated Surveillance Requirements in TS 3.7.10, "Emergency Core Cooling System (ECCS) Pump Room Exhaust Filtration System (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. When the ECCS pumps are operated post-accident, air flow from the ECCS pump room area is 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 analyses under either the current licensing basis or the proposed revision to the licensing basis to implement the AST. Additionally, this system is not a risk significant system as determined by the plant specific probabilistic assessment. As a result, the requirements contained in this TS do not meet any of the four 10 CFR 50.36(c)(2)(ii) criteria for items for which TS LCOs must be established and deletion of this TS is justified.

4.2.5 Remove the TS LCO for Inoperable Conditions Involving SFPEVS Charcoal Adsorber

The proposed revision removes the inoperable Conditions involving SFPEVS charcoal adsorber and deletes the corresponding Surveillance Requirement for filter testing from TS 3.7.11, "Spent Fuel Pool Exhaust Ventilation System (SFPEVS)" and deletes references to SFPEVS filter testing in TS 5.5.11 "Ventilation Filter Testing Program." The SFP charcoal and high efficiency particulate air filters are not credited in the proposed revision to the licensing basis to implement the AST. Additionally, these filters are not risk significant components as determined by the plant specific probabilistic risk assessment. The SFPEVS exhaust fan is the only component credited in the design basis FHA analysis in Section 4.1.2 described above. As a result, the charcoal adsorber requirements contained in this TS do not meet any of the four 10 CFR 50.36(c)(2)(ii) criteria for items for which TS LCOs must be established. Therefore, removal of this requirement is justified.

4.2.6 Revision to Containment Penetration LCO

Technical Specification 3.9.3, Containment Penetrations, specifies the status of each type of containment penetration during core alterations and fuel handling within Containment. Limiting Condition for Operation 3.9.3.d requires penetrations providing direct access from the containment atmosphere to the outside atmosphere to be closed by a manual or automatic isolation valve, blind flange, or equivalent, or is capable of being closed by an operable containment purge valve isolation system. In accordance with TSTF-312 (Reference 6), CCNPP is proposing to add a note under 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. Limiting Condition for Operation 3.6.3, Containment Isolation Valves, ACTION Note 1 allows containment isolation valves to be opened in Modes 1 through 4 under administrative control. In this condition, the accident analyses credit the Containment as a barrier. In the lower energy conditions of LCO 3.9.3, opening containment isolation valves under administrative control is less risk significant. Therefore, this change is proposed to provide a consistent approach to containment boundary issues.

As indicated in Section 4.1.2 above, the proposed FHA analysis in Enclosure (2) assumes that the radioactive release is unfiltered, completely released over a two hour time period, and released with the most limiting dispersion coefficients; therefore, it supports penetration flow paths that have direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative control. Actual offsite doses in the event of a fuel handling incident will be less because containment closure will be established according to the requirements of the administrative control following the accident. In a similar manner, allowing containment penetration flow paths to be opened under administrative control does not negatively impact the dose exposure of the Control Room operators following a fuel handling incident.

Furthermore, the majority of the containment penetrations that would be allowed open under administrative control would open into the Auxiliary Building penetration room, which has a filtered release. Since the Control Room atmospheric dispersion coefficient assumes containment leakage from the ventilation stack and no filtration is assumed prior to reaching the Control Room ventilation system, this analysis bounds all filtered and unfiltered release paths. The filters in the Auxiliary Building penetration room provide an additional conservatism beyond the calculated results.

The following discussion addresses the proposed change with respect to meeting the requirements of the applicable Draft Design Criteria.

Draft Criterion 17 – Monitoring Radioactivity Releases (Category B) states that means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

The proposed Technical Specification (LCO 3.9.3) change will allow for containment penetrations to be opened under administrative control. Means are provided for monitoring the containment atmosphere, which include continuous air monitoring. High radiation inside Containment will also cause the containment purge valves to automatically close and would notify the operators that containment closure is needed. As indicated Section 4.1.2 above, the proposed FHA analysis in Enclosure (2) assumes that the radioactive release is unfiltered and completely released over a two hour time period Therefore, the monitoring of the containment atmosphere provides additional assurance that the actual offsite doses will be bounded by the current analysis.

Draft Criterion 18 – Monitoring Fuel and Waste Storage (Category B) states that monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposure.

Monitoring and alarm instrumentation are provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposure. The proposed change to LCO 3.9.3 has no impact on how the requirements of Draft Criterion 18 are met.

Draft Criterion 70 – Control of Releases of Radioactivity to the Environment (Category B) states that the facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified, (a) on the basis of 10 CFR Part 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur, and (b) on the basis of 10 CFR Part 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

The wording of the draft GDC 70 refers to the operation of the radioactive WPS. Our response to the draft GDC (Reference 7) and the final GDC (GDC 60) refers specifically to radioactive WPSs. The proposed changes have no impact on how the requirements of draft GDC 70 are met.

As stated in our response to draft GDC 70 (Reference 7), the radioactive WPS collects, segregates, processes, and disposes of radioactive solids, liquids, and gases in such a manner as to comply with 10 CFR Part 20.

Solid wastes are processed in a batch manner for off-site disposal. Processed liquid wastes and gaseous wastes released to the environment are monitored and discharged with suitable dilution to assure tolerable activity levels on the site and at the site boundary. Holdup capacity in the reactor coolant WPS is 360,000 gallons; the miscellaneous WPS has a storage capacity of 8,000 gallons. All liquid wastes are sampled to establish their acceptability for release.

The contents of the waste gas decay tanks will be sampled, and a release rate established consistent with the prevailing environmental conditions. A capability is provided for 60-day holdup of waste gas. In-line monitoring will provide a continuous check on the release of activity.

Under incident conditions, radioactive gaseous effluents that may be released into enclosed areas are collected by the ventilation systems and discharged to the plant vent. Permanently installed area radiation detectors and the plant vent radioactivity detectors are used to monitor the discharge levels to the environment. In addition, portable radiation monitors are available on site for supplemental surveys. The releases from these incidents have been calculated to be less than 10 CFR Part 20 and 10 CFR 50.67 guidelines.

4.2.7 Increase in CREVS Flow Rate

The proposed revision increases the CREVS flow rate from 2,000 cfm to 10,000 cfm in TS 5.5.11, "Ventilation Filter Testing Program." This revision is required to meet Control Room dose regulatory limits for the DBAs described in Section 4.1 above. From the TS requirement and safety perspective, this change is more conservative than the existing requirement. A plant modification will be performed to increase the CREVS flow rate, which meets the requirements of Regulatory Guide 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 Generic Letter 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal."

5. <u>REGULATORY ANALYSIS</u>

5.1 No Significant Hazards Consideration

Calvert Cliffs Nuclear Power Plant, Inc. (CCNPP) is proposing an amendment to revise the CCNPP licensing basis to fully implement the Regulatory Guide 1.183 radiological alternate source term (AST). Implementation of AST for CCNPP consists of reevaluation of the following radiological DBAs:

a) Maximum Hypothetical Accident (MHA),
b) Fuel Handling Accident (FHA),
c) Main Steam Line Break (MSLB),
d) Steam Generator Tube Rupture (SGTR),
e) Seized Rotor Event (SRE),
f) Control Element Assembly Ejection Accident (CEAEA),

g) Waste Gas Incident (WGI), and

h) Waste Processing Incident (WPI)

In addition, CCNPP is proposing an amendment to revise the CCNPP Technical Specifications that are associated with and justified by the analyses performed to support AST. The proposed changes have been evaluated against the standards in 10 CFR 50.92 and have been determined to not involve a significant hazards consideration in that:

1. *Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.*

   The results of the applicable radiological design basis accidents (DBAs) re-evaluation demonstrated that, with the requested changes, the dose consequences of these limiting events are within the regulatory limits and guidance provided by the Nuclear Regulatory Commission in 10 CFR 50.67 and Regulatory Guide 1.183 for AST methodology. The AST is an input to calculations used to evaluate the consequences of an accident and does not by itself affect the plant response or the actual pathway of the activity released from the fuel. It does, however, better represent the physical characteristics of the release such that appropriate mitigation techniques may be applied.

   The change from the original source term to the new proposed AST is a change in the analysis method and assumptions and has no effect on accident initiators or causal factors that contribute to the probability of occurrence of previously analyzed accidents. Use of an AST to analyze the dose effect of DBAs shows that regulatory acceptance criteria for the new methodology continues to be met. Changing the analysis methodology does not change the sequence or progression of the accident scenario.

   The proposed Technical Specification changes reflect the plant configuration that will either support implementation of the AST analyses or eliminate requirements that are no longer needed as a result of the revised DBA analyses. The equipment affected by the proposed changes is mitigative in nature and relied upon after an accident has been initiated. The operation of various filtration systems have been considered in the evaluations for these proposed changes. While the operation of some systems does change with the implementation of an AST, the affected systems are not accident initiators; and application of the AST methodology, itself, is not an initiator of a DBA.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.*

   As described in Item 1 above, the changes proposed in this license amendment request involve the use of a new analysis methodology and related regulatory acceptance criteria. The proposed Technical Specification changes reflect the plant configuration that will either support implementation of the new methodology or eliminate requirements that are no longer needed as a result of the new methodology. No new or different accidents result from utilizing the proposed changes. Although the proposed changes require modification to the Control Room emergency ventilation system and installation of automatic isolation dampers and radiation monitors at Access Control HVAC Unit RTU-1 and Access Control Air Conditioning Unit 13 on the Auxiliary Building roof, none of these changes can initiate a new or different kind of accident since they are only related to system capabilities that provide protection from accidents that have already occurred. As a result, no new failure modes are being introduced that could lead to different accidents. These changes do not alter the nature of events postulated in the Updated Final Safety Analysis Report nor do they introduce any unique precursor mechanisms.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. *Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.*

   As described in Item 1 above, the changes proposed in this license amendment request involve the use of a new analysis methodology and related regulatory acceptance criteria. The proposed Technical Specification changes reflect the plant configuration that will either support implementation of the new methodology or eliminate requirements that are no longer needed as a result of the new methodology. Safety margins and analytical conservatisms have been evaluated and have been found acceptable. The analyzed events have been carefully selected and, with plant modification, margin has been retained to ensure that the analyses adequately bound postulated event scenarios. The analyses have been performed using conservative methodologies, as specified in Regulatory Guide 1.183. The dose consequences of these DBAs remain within the acceptance criteria presented in 10 CFR 50.67, "Accident Source Term," and Regulatory Guide 1.183. The proposed changes continue to ensure that the doses at the exclusion area boundary and low population zone boundary, as well as the Control Room, are within corresponding regulatory limits.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above evaluation, Calvert Cliffs has concluded that the proposed amendment involves no significant hazards considerations under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Regulatory Requirements

The proposed changes have been evaluated to determine compliance with applicable regulatory requirements.

The revised analyses for the DBAs identified in Section 4 are based on 10 CFR 50.67 and use the regulatory guidance of Regulatory Guide 1.183. The analyses demonstrate compliance with these regulatory guides and criteria. Use of the new analysis method replaces 10 CFR Part 100 as the applicable dose acceptance criteria for all DBAs.

General Design Criterion 19 requires that holders of an operating license using an AST under 10 CFR 50.67 shall meet the requirements of the criterion by ensuring the radiation exposures to Control Room occupants shall not exceed 5 rem TEDE dose. The analysis provided to support the requested changes demonstrates that this requirement is met.

Calvert Cliffs has determined that the proposed changes do not require any exemptions or relief from regulatory requirements. The proposed use of an AST to evaluate the consequences of a DBA results in a change to the existing licensing basis analysis described in the Updated Final Safety Analysis Report. In accordance with 10 CFR 50.71, Calvert Cliffs will update the Updated Final Safety Analysis Report to reflect the proposed new analysis method. The changes to the TSs incorporate assumptions used in the new analysis.

Compliance with draft GDC 17, 18, and 70 is described in Section 4.2.6.

6. ENVIRONMENTAL CONSIDERATION

Adoption of the AST and associated TS changes, which implement certain conservative assumptions in the AST analyses, will not result in physical changes to the plant that could significantly alter the type or amounts of effluents that may be released offsite. No changes to operational parameters that could affect effluent releases have been proposed.

The implementation of the AST has been evaluated in revisions to the analyses of the limiting DBAs at Calvert Cliffs Nuclear Power Plant. Based upon the results of these analyses, it has been demonstrated that with the proposed change, the dose consequences of this limiting event are within the regulatory requirements specified by the NRC for use with the AST (i.e., 10 CFR 50.67 and 10 CFR Part 50, Appendix A, GDC 19). Thus, there will be no significant increase in either individual or cumulative occupational radiation exposure.

We have determined that operation with the proposed amendment would not result in any significant change in the types, or significant increases in the amounts, of any effluents that may be released offsite, nor would it result in any significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed amendment.

7. PRECEDENTS

The NRC has previously approved implementation of the AST methodology at a number of other nuclear power stations including:

- Surry Power Station in Amendment No. 230, dated March 8, 2002;
- Kewaunee Nuclear Power Plant in Amendment No. 166, dated March 17, 2003; and
- H. B. Robinson Steam Electric Plant in Amendment No. 201, dated September 24, 2004.

8. REFERENCES

(1) Letter from Mr. G. Vanderheyden (CCNPP) to Document Control Desk (NRC), dated November 23, 2004, Supplemental Response to NRC Generic Letter 2003-01, "Control Room Habitability"

(2) Letter from Mr. G. Vanderheyden (CCNPP) to Document Control Desk (NRC), dated December 5, 2003, Response to NRC Generic Letter 2003-01, "Control Room Habitability"

(3) NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000

(4) J.J. DiNunno et al., Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," U.S. Atomic Energy Commission (now USNRC), 1962

(5) L. Soffer et al., "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, USNRC, February 1995

(6) Industry/TSTF Standard Technical Specification Change Traveler TSTF-312, Administratively Control Containment Penetrations, Revision 1

(7) Letter from Mr. J. W. Gore, Jr. (BG&E) to Dr. P. A. Morris (US AEC), dated January 4, 1971, Amendment No. 11

9. REGULATORY COMMITMENTS

1. Calvert Cliffs Nuclear Power Plant, Inc. will perform the required plant modification to increase the CREVS flow rate from 2,000 cfm to 10,000 cfm in support of the proposed revision to the licensing basis to implement AST and revision to the TS 5.5.11.
2. Calvert Cliffs Nuclear Power Plant, Inc. will install automatic isolation dampers and radiation monitors at Access Control HVAC Unit RTU-1 and Access Control Air Conditioning Unit 13 on the Auxiliary Building roof in support of the proposed revision the licensing basis to implement AST.
3. The Technical Specification Bases and the UFSAR descriptions will be revised as appropriate to support the proposed license amendment request for the implementation of the AST and revision of the associated revision to TS.
4. The reactor coolant waste evaporator will be retired-in-place.

10. ENCLOSURES

1. CA06449 MHA Radiological Consequences Design Basis Calculation Using AST
2. CA06450 FHA Radiological Consequences Design Basis Calculation Using AST
3. CA06452 MSLB Radiological Consequences Design Basis Calculation Using AST
4. CA06453 SGTR Radiological Consequences Design Basis Calculation Using AST
5. CA06451 SRE Radiological Consequences Design Basis Calculation Using AST
6. CA06454 CEAEA Radiological Consequences Design Basis Calculation Using AST
7. CA06604 WGI Radiological Consequences Design Basis Calculation Using AST
8. CA06608 WPI Radiological Consequences Design Basis Calculation Using AST
9. CA06012 Atmospheric Dispersions Coefficient (X/Q) Calculation
10. CA06358 Source Terms Calculation
11. CA06421 Gas Gap Isotopic Fraction Calculation
12. CA06422 Primary and Secondary Isotopic Calculations
13. Compact Disk Containing Input Data for DBA Calculations