March 6, 2008

Mr. James J. Sheppard
President and Chief Executive Officer
STP Nuclear Operating Company
South Texas Project Electric
Generating Station
P.O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: ADOPTION OF ALTERNATE RADIOLOGICAL SOURCE TERM IN ASSESSMENT OF DESIGN-BASIS ACCIDENT DOSE CONSEQUENCES (TAC NOS. MD4996 AND MD4997)

Dear Mr. Sheppard:

The Commission has issued the enclosed Amendment No. 182 to Facility Operating License No. NPF-76 and Amendment No. 169 to Facility Operating License No. NPF-80 for the South Texas Project, Units 1 and 2, respectively. The amendments consist of changes to the licensing basis and Technical Specifications (TS) in response to your application dated March 22, 2007, as supplemented by letters dated April 10, July 18, October 11, November 13, December 13, and December 18, 2007.

The amendments revise the licensing basis, pursuant to Title 10 of the *Code of Federal Regulations,* Section 50.67, "Accident Source Term," and approve the methodology for evaluating radiological consequences of design-basis accidents as described in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents (DBAs) at Nuclear Power Reactors." The amendments revise the TS in support of the revisions to the licensing basis.

A copy of our related Safety Evaluation is also enclosed.

Sincerely,

/RA/

Mohan C. Thadani, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosures:
1. Amendment No. 182 to NPF-76
2. Amendment No. 169 to NPF-80
3. Safety Evaluation

cc w/encls: See next page

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NAME	MThadani	JBurkhardt	AHiser (*)	MHart (*)	RDennig (*)	GWaig:TKolb for (**)	MSpencer (**)	THiltz
DATE	3/6/08	3/6/08	9/6/07	11/15/07	12/20/07	2/15/08	2/28/08	3/6/08

OFFICIAL RECORD COPY (*) SE input memo, (**) See previous concurrence

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STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 182
License No. NPF-76

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

   A. The application for amendment by STP Nuclear Operating Company (STPNOC)* acting on behalf of itself and for NRG South Texas LP, the City Public Service Board of San Antonio (CPS), and the City of Austin, Texas (COA) (the licensees), dated March 22, 2007, as supplemented by letters dated April 10, July 18, October 11, November 13, December 13, and December 18, 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, as amended, the provisions of the Act, and the rules and regulations of the Commission;

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

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

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

---

* STPNOC is authorized to act for NRG South Texas LP, the City Public Service Board of San Antonio, and the City of Austin, Texas, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-76 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 182, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. STPNOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 120 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Thomas G. Hiltz, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility
Operating License No. NPF-76
and the Technical Specifications

Date of Issuance: March 6, 2008

STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-499

SOUTH TEXAS PROJECT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 169
License No. NPF-80

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

   A. The application for amendment by STP Nuclear Operating Company (STPNOC)* acting on behalf of itself and for NRG South Texas LP, the City Public Service Board of San Antonio (CPS), and the City of Austin, Texas (COA) (the licensees), dated March 22, 2007, as supplemented by letters dated April 10, July 18, October 11, November 13, December 13, and December 18, 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, as amended, the provisions of the Act, and the rules and regulations of the Commission;

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

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

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

* STPNOC is authorized to act for NRG South Texas LP, the City Public Service Board of San Antonio, and the City of Austin, Texas, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-80 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 169, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. STPNOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 120 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Thomas G. Hiltz, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility
Operating License No. NPF-80
and the Technical Specifications

Date of Issuance: March 6, 2008

ATTACHMENT TO LICENSE AMENDMENT NOS. 182 AND 169

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

DOCKET NOS. 50-498 AND 50-499

Replace the following pages of the Facility Operating License Nos. NPF-76 and NPF-80, and 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.

Facility Operating License No. NPF-76

REMOVE	INSERT
-4-	-4-

Facility Operating License No. NPF-80

REMOVE	INSERT
-4-	-4-

Technical Specifications

REMOVE	INSERT	REMOVE	INSERT
ix	ix	3/4 8-9a	3/4 8-9a
x	x	3/4 8-13	3/4 8-13
xi	xi	3/4 8-13a	3/4 8-13a
1-2	1-2	3/4 8-16	3/4 8-16
3/4 3-20	3/4 3-20	3/4 9-3	3/4 9-3
3/4 3-25	3/4 3-25	3/4 9-4	--
3/4 3-26	3/4 3-26	3/4 9-5	--
3/4 3-27	3/4 3-27	3/4 9-6	--
3/4 3-28	3/4 3-28	3/4 9-7	--
3/4 3-34	3/4 3-34	3/4 9-10	3/4 9-10
3/4 3-35	3/4 3-35	3/4 9-14	3/4 9-14
3/4 3-44	3/4 3-44	3/4 9-15	--
3/4 3-48	3/4 3-48	3/4 9-16	--
3/4 3-49	3/4 3-49		
3/4 7-16	3/4 7-16		
3/4 7-18	3/4 7-18		
3/4 7-19	3/4 7-19		
3/4 7-20	--		
3/4 7-21	--		

SAFETY EVALUATION BY OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 182 AND 169

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

STP NUCLEAR OPERATING COMPANY, ET AL.

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

1.0 INTRODUCTION

By letter dated March 22, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML070890474), as supplemented by letters dated April 10, July 18, October 11, November 13, December 13, and December 18, 2007 (ADAMS Accession Nos. ML071620446, ML072050341, ML072970308, ML073250369, ML073580100, and ML073610144, respectively), STP Nuclear Operating Company (STPNOC, the licensee) submitted a license amendment request (LAR) to amend the South Texas Project (STP), Units 1 and 2 licensing basis and Technical Specifications (TS). Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 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 [DBAs] at Nuclear Power Reactors," the licensee performed a revised analysis of consequences of DBAs utilizing the alternate source term (AST) methodology described in RG 1.183, with the exception that the methodology for evaluation of equipment qualification was unchanged. The licensee requested approval of the revised analysis of radiological consequences of DBAs.

Additionally, the licensee requested approval of changes to TS in support of the revised licensing basis for radiological consequences of DBAs implementing a full-scope AST at STP, Units 1 and 2.

The supplemental letters dated April 10, July 18, October 11, November 13, December 13, and December 18, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on July 31, 2007 (72 FR 41788).

The U.S. Nuclear Regulatory Commission (NRC) staff has evaluated STPNOC's LAR. The NRC staff's evaluation follows.

2.0 REGULATORY EVALUATION

The NRC staff evaluated the licensee's analysis of radiological consequences of affected DBAs for implementation of the AST methodology at STP, Units 1 and 2 against the radiological dose guidelines specified in 10 CFR 50.67(b)(2), and dose limits specified in 10 CFR 50, Appendix A, General Design Criterion (GDC) 19. The dose guidelines in 10 CFR 50.67(b)(2) are: 25 roentgen equivalent man (rem) total effective dose equivalent (TEDE) at the exclusion area boundary (EAB) for any 2-hour period following the onset of the postulated fission product release; 25 rem TEDE to an individual located at any point on the outer boundary of the low population zone (LPZ), who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage); and the dose limit of 5 rem TEDE to the occupants of the control room (CR) for access and occupancy of the CR for the duration of the postulated fission product release. The use of 25 rem TEDE is not intended to imply that this value constitutes an acceptable limit for emergency doses to the public under accident conditions. Rather, the 25 rem TEDE value has been stated in this section as a reference value, which can be used in evaluation of proposed design-basis changes with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation.

This safety evaluation (SE) addresses the impact of the proposed changes on previously analyzed DBA radiological consequences and the acceptability of the revised analysis results. The regulatory requirements from which the NRC staff based its acceptance are the accident dose reference values in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 of RG 1.183 and Standard Review Plan (SRP) Section 15.0.1. The licensee has not proposed any significant deviation or departure from the guidance provided in RG 1.183. Also, the licensee proposed changes to the TS in support of the implementation of proposed changes to the analysis of radiological consequences of DBAs.

The NRC staff's evaluation is based upon the following regulations, regulatory codes, guides, and standards:

- 10 CFR 50.36, "Technical Specifications."
- 10 CFR 50.67, "Accident Source Term."
- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 19, "Control room."
- RG 1.23, "Onsite Meteorological Programs," Rev. 0, February 1972.
- 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. 3, June 2001.
- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Rev. 1, November 1982.

- RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” Rev. 0, July 2000.
- RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” Rev. 0, June 2003.
- RG 1.196, “Control Room Habitability at Light-Water Nuclear Power Reactors,” Rev. 0, May 2003.
- NUREG-0409, ”Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident,” May 1985.
- NUREG-0800, “Standard Review Plan,” Section 2.3.4, “Short-Term Dispersion Estimates for Accident Releases,” Rev. 3, March 2007.
- NUREG-0800, “Standard Review Plan,” Section 6.4, “Control Room Habitability System,” Rev. 3, March 2007.
- NUREG-0800, “Standard Review Plan,” Section 6.5.2, “Containment Spray as a Fission Product Cleanup System,” Rev. 4, March 2007.
- NUREG-0800, “Standard Review Plan,” Section 15.0.1, “Radiological Consequence Analyses Using Alternative Source Terms,” Rev. 0, July 2000.
- NUREG/CR-5950, “Iodine Evolution and pH [Potential of Hydrogen] control,” December 1992.
- NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” February 1995.
- NUREG-0737, “Clarification of TMI [Three Mile Island] Action Plan Requirements,” November 1980.
- Technical Specification Task Force (TSTF)-51, Revision 2, “Revised containment requirements during handling irradiated fuel and core alterations.”

3.0 TECHNICAL EVALUATION

3.1 Radiological Consequences of DBAs

As stated in RG 1.183, Regulatory Position 5.2, the DBAs addressed in the appendices of RG 1.183 were selected from accidents that may involve damage to irradiated fuel. RG 1.183 does not address DBAs with radiological consequences based on TS for reactor or secondary coolant-specific activities only. The inclusion or exclusion of a particular DBA in RG 1.183 should not be interpreted as indicating that an analysis of that DBA is required or not required.

The licensees should analyze the DBAs that are affected by the plant-specific proposed applications of an AST.

The licensee performed analyses for the full implementation of the AST, in accordance with the guidance in RG 1.183, and Section 15.0.1 of the SRP. Also, the licensee's AST analyses were based on the pressurized-water reactor (PWR) DBAs identified in RG 1.183 that could potentially result in significant CR and offsite doses. They include the loss-of-coolant accident (LOCA), the fuel handing accident (FHA), the main steamline break accident (MSLB), the steam generator tube rupture accident (SGTR), the control rod ejection accident (CREA), and the locked rotor accident (LRA).

In its LAR, the licensee stated that the AST methodology is being used to resolve a non-conforming condition at STP where testing resulted in CR unfiltered in-leakage at a value greater than the current accident dose analysis assumption.

The licensee plans to submit a revision to TS 3/4.7.7, "Control Room Makeup and Cleanup Filtration System," in a separate licensing action. This revision will be based on TS Task Force (TSTF) Traveler-448, Revision 3, which includes a surveillance program for measuring CR in-leakage. The in-leakage acceptance criterion in the surveillance will be the revised 100 cubic feet per minute (cfm) value assumed in the current DBA analyses supporting the AST amendments.

The licensee identified the following condition related to this LAR. Westinghouse Electric Company Nuclear Safety Advisory Letter (NSAL)-06-15, dated December 13, 2006, advised operators of Westinghouse plants that the single-failure scenario for the SGTR analysis that licensees used in their accident analysis may not be limiting. The licensee has evaluated the applicability of NSAL-06-15 against the accident analysis assumptions and has determined that the current single-failure assumption for the STP SGTR analysis is not limiting. Therefore, the licensee is operating under compensatory measures to meet regulatory dose guidelines. The licensee plans to resolve this condition at the earliest opportunity so that the assumptions, including the limiting single failure, for the SGTR accident analysis described herein are consistent with the plant response to this event. To support the limiting single-failure assumptions in the SGTR analysis, STP will maintain an administrative limit for reactor coolant system (RCS) dose equivalent iodine 131 (DEI) so that the radiological dose reference values for the SGTR analysis remain bounding, and the licensee will continue to comply with GDC 19.

The licensee provided additional information describing the potential CR and offsite dose consequences for an SGTR accident assuming that the RCS specific activity was at the TS 3.4.8, "RCS Specific Activity" limiting condition for operation (LCO) and with a failed open main steam isolation valve (MSIV) as the limiting single-failure assumption. The evaluation showed that the offsite dose consequences would meet the applicable guidelines. The analysis also demonstrated to the satisfaction of the staff that, with the administrative limit for RCS DEI, the CR dose would be reduced to a value well within the applicable limits. In addition, the licensee stated that final approval of the modification and issuance of the design change package for Unit 1 is planned for late 2007 and for Unit 2 is planned for spring 2008. The modification is planned to be implemented in Unit 1 during the 2009 fall outage and in Unit 2 during the 2008 fall outage. In any event, the licensee must comply with the GDC 19 CR dose

limit, which is equal to the CR dose limit in 10 CFR 50.67(b)(2)(iii). Therefore, the staff finds that the licensee's approach to maintain RCS DEI at an appropriate administrative limit until plant modifications are completed is acceptable for mitigation of a CR unfiltered in-leakage non-conforming condition.

Each DBA radiological source term used in the AST analyses was developed based on a core power level of 4100 megawatts thermal (MWt). The core power level used to determine the RCS steam releases is 3876 MWt which represents the licensed power of 3853 MWt with a 0.6 percent increase to account for measurement uncertainties. The use of 4100 MWt for the AST DBA radiological source term analyses bounds the current licensed core thermal power level of 3853 MWt and is, therefore, acceptable to the staff for use in the full implementation of the AST at STP, Units 1 and 2.

The licensee has performed its evaluation based on full implementation of the AST as defined in RG 1.183, except that the licensee has determined that the current Technical Information Document (TID) 14844, Atomic Energy Commission (AEC), 1962, "Calculation of Distance Factors for Power and Test Reactors Sites," accident source term will remain the licensing basis for equipment qualification (EQ).

Regulatory Position 6 of RG 1.183 states that the NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted and that until such time as this generic issue is resolved, licensees may use either the AST or the TID-14844 assumptions for performing the required EQ analyses. This issue has been resolved as documented in a memorandum dated April 30, 2001 (ADAMS Accession No. ML011210348), and in NUREG-0933, Supplement 25, June 2001 (ADAMS Accession No. ML012190402). The conclusion of Generic Issue 187 states the following: "The staff concluded that there was no clear basis for back-fitting the requirement to modify the design basis for equipment qualification to adopt the AST. There would be no discernible risk reduction associated with such a requirement. Licensees should be aware, however, that a more realistic source term would potentially involve a larger dose for equipment exposed to sump water for long periods of time. Longer term equipment operability issues associated with severe fuel damage accidents, (with which the AST is associated) could also be addressed under accident management or plant recovery actions as necessary." Therefore, in consideration of the above-cited references, the staff finds that it is acceptable for the TID-14844 accident source term to remain the licensing basis for EQ at STP, Units 1 and 2.

RG 1.183, Regulatory Position 4.3, under Other Dose Consequences, states that: "[t]he guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737. Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE."

As part of the DBA LOCA analysis, the licensee evaluated the radiation levels from contained sources, the containment structure, the CR, and the Technical Support Center (TSC) filters. The licensee used these evaluations to determine if an impact on the following areas covered

by NUREG-0737, “Clarification of TMI Action Plan Requirements,” November 1980, would occur as a result of any increases in the associated radiation levels:

- Current licensing basis (CLB) radiological dose analyses for post-accident vital area access and post-accident sampling as described in NUREG-0737, Item II.B.2 and Item II.B.3,

- CLB radiological dose analyses for the post-accident containment high-range radiation monitors, as described in NUREG-0737, Item II.F.1, and

- CLB control room post-accident radiological dose analyses for emergency support facility upgrades and control room habitability as described in NUREG-0737, Items III.A.1.2 and III.D.3.4.

The licensee has determined that the CLB calculations used in support of post-accident vital area access are unaffected based on an assessment of AST versus TID-14844 contained sources. The licensee has determined that the current calculated doses, based on the TID-14844 source term, bound the corresponding doses that would be calculated based on the AST. In addition, the licensee has not credited any new operator missions to support the implementation of the AST methodology.

The requirements of NUREG-0737 for Post Accident Sampling System (PASS) were deleted as part of Amendment No. 133 issued November 7, 2001 (ADAMS Accession No. ML013250294). The CLB analysis for the containment high-range radiation monitors used to monitor post-accident primary containment radiation levels is based on a source term which differs from either TID-14844 or the AST. Therefore, there is no impact of AST implementation on the CLB containment high-range monitor evaluation. The licensee provided full dose consequence analyses for both the CR and TSC as part of its AST LAR. Therefore, the staff finds that the licensee fully addressed the issue of maintaining consistency with the NUREG-0737 evaluations while incorporating the AST into the plant licensing basis for dose consequence analyses.

A full implementation of the AST is proposed for STP, Units 1 and 2. Therefore, to support the licensing and plant operation changes discussed in the LAR, the licensee analyzed the following accidents employing the AST as described in RG 1.183.

1. Loss-of-coolant Accident (LOCA)

2. Fuel Handling Accident (FHA)

3. Main Steam Line Break (MSLB) Accident

4. Steam Generator Tube Rupture (SGTR) Accident

5. Control Rod Ejection Accident (CREA)

6. Locked Rotor Accident (LRA)

The DBA dose consequence analyses evaluated the integrated TEDE dose at the EAB for the worst 2-hour period following the onset of the accident. The integrated TEDE doses at the outer boundary of the LPZ during the entire period of the passage of the radioactive cloud resulting from postulated release of fission products, and the integrated dose to an STP, Units 1 and 2 CR operator were evaluated for the duration of the accident. The dose consequence analyses were performed by the licensee using the "RADTRAD: Simplified Model for RADionuclide Transport and Removal and Dose Estimation," Version 3.03, computer code. The development of the RADTRAD radiological consequence computer code was sponsored by the NRC, as described in NUREG/CR-6604, and was developed by Sandia National Laboratories for the NRC. The code estimates transport and removal of radionuclides and radiological consequence doses at selected receptors. The staff performs independent confirmatory dose evaluations using the RADTRAD computer code. The results of the evaluations performed by the licensee, as well as the applicable dose acceptance guidelines from RG 1.183, are shown in Table 1 of this SE.

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."

In accordance with RG 1.183, the licensee generated the core and worst case fuel assembly radionuclide inventories for use in determining source term inventories using the ORIGEN code version 2.1. The licensee assumed a period of irradiation that was sufficient to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The licensee modeled the core as a three-region equilibrium cycle core at end of life with the three regions having operated at 39.31 megawatts days per metric ton of uranium (MWD/MTU) for 509, 1018, and 1527 effective full power days (EFPD), respectively. The inventories, consisting of the curie levels for 63 dose significant isotopes at end of fuel cycle, formed the source term input for the RADTRAD dose evaluations.

As stated in RG 1.183, the release fractions associated with the light-water reactor (LWR) core inventory released into containment for the DBA 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 MWD/MTU of uranium provided that the maximum linear heat generation rate does not exceed 6.3 kilowatts per foot (kW/ft) for peak rod average power for burnups exceeding 54,000 MWD/MTU.

The licensee performed an evaluation to determine the best estimate fuel rod average burnup and power for typical 18-month cycle designs. The licensee's evaluation indicates that the RG 1.183 limitation of 6.3 kW/ft peak rod average power for burnups exceeding 54,000 MWD/MTU for the application of the AST is met with significant margin. The licensee

has determined for burnups exceeding 54,000 MWD/MTU, the highest corresponding linear heat rate is 5.7 kW/ft.

The licensee used committed effective dose equivalent (CEDE) and effective dose equivalent (EDE) DCFs from Federal Guidance Reports (FGR) 11 and 12 to determine the TEDE dose in accordance with AST evaluations. The use of ORIGEN and DCFs from FGR-11 and FGR-12 is in accordance with RG 1.183 guidance and is, therefore, acceptable to the staff.

3.1.1 Loss-of-Coolant Accident (LOCA)

The design-basis LOCA radiological consequence LOCA analysis is a deterministic evaluation based on the assumption of a major rupture of the primary RCS piping. The accident scenario assumes the deterministic failure of the ECCS to provide adequate core cooling which results in a significant amount of core damage as specified in RG 1.183. This general scenario does not represent any specific accident sequence, but is representative of a class of severe damage incidents that were evaluated in the development of the RG 1.183 source term characteristics. Such a scenario would be expected to require multiple failures of systems and equipment and lies beyond the likely incidents evaluated for design-basis transient analyses.

The LOCA considered in this evaluation is a complete and instantaneous circumferential severance of the primary RCS piping, which would result in the maximum fuel temperature and primary containment pressure among the full range of LOCAs. Due to the postulated loss of core cooling, the fuel heats up, resulting in the release of fission products. The fission product release is assumed to occur in phases over a 2-hour period.

When using the AST for the evaluation of a design-basis LOCA for a PWR, it is assumed that the initial fission product release to the containment will last for 30 seconds and will consist of the radioactive materials dissolved or suspended in the RCS liquid. After 30 seconds, fuel damage is assumed to begin and is characterized by clad damage that releases the fission product inventory assumed to reside in the fuel gap. The fuel gap release phase is assumed to continue until 30 minutes after the initial breach of the RCS. As core damage continues, the gap release phase ends and the early in-vessel release phase begins. The early in-vessel release phase continues for the next 1.3 hours. The licensee used the LOCA source term release fractions, timing characteristics, and radionuclide grouping as specified in RG 1.183 for evaluation of the AST.

In the evaluation of the LOCA design-basis radiological analysis the licensee considered dose contributions from the following potential activity release pathways:

- Containment leakage directly to the atmosphere
- Containment leakage into the electrical penetration room
- Engineered safety feature (ESF) system leakage
- RCS releases through the containment supplemental purge system prior to isolation

The licensee considered the following DBA LOCA dose contributors to the control room habitability envelope (CRHE) analysis:

- Contamination of the CR atmosphere by released activity
- Plume shine from released activity
- Shine from the containment, the electrical penetration area and CR filter loading

3.1.1.1 Source Term

The licensee followed all aspects of the guidance outlined in RG 1.183, Regulatory Position 3, regarding the core inventory, release fractions and timing for the evaluation of the LOCA.

The LOCA analysis assumes that iodine will be removed from the containment atmosphere by both containment sprays and natural diffusion to the containment walls. As a result of these removal mechanisms a large fraction of the released activity will be deposited in the containment sump. The sump water will retain soluble gases and soluble fission products such as iodines and cesium, but not noble gases. The guidance from RG 1.183 specifies that the iodine deposited in the sump water can be assumed to remain in solution as long as the containment sump pH is maintained at or above 7.

The licensee conducted an evaluation of containment sump pH in order to ensure that particulate iodine deposited into the containment sump water does not re-evolve beyond the amount recognized in the DBA LOCA analysis. The licensee's determination of pH was performed using the methodology outlined in NUREG/CR-5950, "Iodine Evolution and pH Control," December 1992, and NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995. The licensee also performed an analysis of the associated iodine decontamination factor (DF) for containment iodine removal and retention.

The design inputs for calculating the containment sump pool pH were conservatively established by the licensee to maximize the acidic contribution to sump pH and minimize the basic contribution. The licensee's pH analysis credits the buffering effect of trisodium phosphate (TSP), which is stored in baskets in the containment sump. The licensee credits the post-LOCA dissolution of the TSP in the containment sump water resulting from the released reactor coolant and injected spray water coming in contact with the stored TSP in the lower elevation of containment. The baskets of TSP are assumed to be submerged during a DBA LOCA thereby buffering the sump pH against LOCA-induced acidity. Contributors to acidity in the sump pool at STP include: boric acid from the reactor coolant system, the accumulators, and the refueling water storage tank (RWST); nitric acid from radiolysis of water; and, hydrochloric acid from radiolysis of chloride-bearing cable jacket insulation.

The licensee made conservative assumptions regarding the amount of cable insulation present and included a factor to account for the addition of more cable in the future. In the containment sump pH analysis, the licensee did not credit the basic alkali metal compounds and cesium compounds that result from fission products co-released with radioactive iodine.

To add to the conservatism of the pH analysis, the licensee assumed that 10 percent of non-noble gas activity remains airborne in the containment for the entire duration of the accident evaluation, even in the presence of sprays. In addition, the licensee assumed that all of the noble gas activity remains airborne in the containment for the duration of the accident. These assumptions result in an increase in the amount of radiation exposure to cables which subsequently results in a higher production of hydrochloric acid due to radiolysis of the cable jacket insulation.

The licensee's analysis credits a reduction of a factor of two for beta shielding of cable in trays to account for the layering of cables which provides a significant amount of self shielding. A beta shielding factor of ten is credited for the 16 percent of cable that is estimated to be in conduit.

The licensee has determined that the buffering effect of TSP is sufficient to maintain the sump pH at or above 7 for the first day following a DBA LOCA. However, after the initial 24 hours following the LOCA, the licensee's evaluation indicates that the sump pH begins to slowly decrease reaching a lower limit of approximately 6.8 by the end of the 30-day duration of the analysis. The RG 1.183 position on postulated iodine re-evolution is based on the sump pH being maintained at or above 7. Therefore, with the pH dropping below 7 during the DBA LOCA, the licensee determined the extent to which iodine re-evolution would increase in response to this effect.

Iodine re-evolution is enhanced by higher temperatures and lower levels of pH in the sump water. To conservatively bound the dose consequence from iodine re-evolution, the licensee assumed that the maximum sump temperature of 266 degrees Fahrenheit (ºF) and the minimum pH value of 6.8 would persist for the entire 30-day duration of the accident. To account for the drop in pH below 7 and the subsequent potential increase in iodine re-evolution, the licensee limited the assumed DF for elemental iodine for spray and natural removal mechanisms inside containment to a value of 60.

The licensee conservatively applied a DF of 60 which corresponds to a pH of 6.8 in the dose analysis, even though the calculated value of pH after 30 days is just below 6.85. This approach is conservative in that the highest sump temperature is used and the lowest pH is assumed throughout the duration of the accident. The licensee asserts that in reality, the assumed DF of 60 should be exceeded at all times since early in the accident the sump pH is greater than 6.8 and later in the accident the sump temperature is much less than the maximum value of 266 ºF.

The licensee's conservative evaluation indicates that the sump pH is transient and would range from 7 at the onset of the DBA LOCA, to approximately 6.8 at the end of the 30-day duration of the accident. The assumed corresponding DF would also be transient and vary from a value of 150 at the onset of the event to a value of 60 at the end of the DBA LOCA analysis. For conservatism the licensee used a DF of 60 for the entire 30-day DBA LOCA analysis to calculate the radiological doses at EAB, LPZ, and in the CR. The staff finds that the licensee used conservative assumptions to bound the radiological consequences of the potential

re-evolution of iodine from the containment sump and that the analysis is acceptable for use in the AST LOCA analysis.

3.1.1.2 Assumptions on Transport in the Primary Containment

3.1.1.2.1 Containment Mixing, Natural Deposition and Leak Rate

In accordance with RG 1.183, the licensee assumed that the activity released from the fuel is mixed instantaneously and homogeneously throughout the free air volume of the containment. The licensee used the core release fractions and timing as specified in RG 1.183 with the termination of the release into containment set at the end of the early in-vessel phase.

The licensee credited the reduction of airborne radioactivity in the containment by natural deposition. The licensee credited an elemental iodine natural deposition removal coefficient of 4.5 per hour until an elemental iodine DF of 60 is reached at 1.855 hours post-LOCA. The licensee did not credit the removal of organic iodine by natural deposition. The licensee applied the elemental iodine natural deposition removal coefficient of 4.5 per hour to both the sprayed and unsprayed volume of the containment. The application of the same removal coefficient for both volumes is based on the assumption that both volumes have the same surface area to volume ratio. The licensee provided additional information to explain the difference in the application of the natural deposition removal coefficient of 4.5 per hour to both the sprayed and unsprayed volume of the containment as opposed to the CLB which uses 3.59 per hour for the sprayed region and 0.91 per hour for the unsprayed region. The licensee stated that the use of the 4.5 per hour for both volumes represents a correction to a conservative conceptual error in the CLB. A stated assumption was made in the CLB that the area-to-volume ratio is the same for both the sprayed and unsprayed regions. However, the resulting 4.5 per hour value was then incorrectly apportioned between the sprayed and unsprayed regions according to the volume fraction. Since the removal lambda is simply the product of the given deposition velocity, and the area-to-volume ratio, the removal rate must be the same for both volumes if the area-to-volume ratio is the same. Therefore, the licensee asserts, and the staff agrees, that the use of the 4.5 per hour value represents a corrected application of the CLB.

RG 1.183, Regulatory Position 3.7 states that, "[t]he primary containment should be assumed to leak at the peak pressure TS leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50 percent of the TS leak rate." Accordingly, the licensee assumed a containment leak rate of 0.30 percent per day for the first 24 hours, after which the containment leak rate is reduced to 0.15 per day for the duration of the accident.

3.1.1.2.2 Containment Spray Assumptions

RG 1.183, Appendix A, Regulatory Position 3.3 states that, "[t]he containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90 percent of the volume and if adequate mixing of unsprayed compartments can be shown." In addition, SRP Section 6.5.2, III,1,c states, "[t]he containment building atmosphere may be considered a single, well-mixed space if the spray covers regions comprising at least 90 percent of the containment building space and if a ventilation system is available for adequate mixing of any unsprayed compartments."

For STP, Units 1 and 2, the volume of the sprayed region is 2.7E+06 ft^3 and the volume of the unsprayed region is 6.8E+05 ft^3. Since the sprayed region represents approximately 80 percent of the total containment volume the licensee used a two volume model to represent the sprayed and unsprayed regions of the containment.

The licensee credited three out of the six Reactor Containment Fan Cooler (RCFC) units for mixing between the sprayed and unsprayed regions of the containment in the AST LOCA analysis. Two of the RCFC units are assumed to be inoperable due to the assumption of the failure of a standby diesel generator to start upon a loss-of-offsite power (LOOP). An additional RCFC unit is assumed to be out-of-service for maintenance at the time of the LOCA.

The licensee only considered the forced convection induced by the RCFC units to model the transfer rate between the sprayed and unsprayed containment regions. For conservatism, the licensee did not credit the effects of natural convection, steam condensation, and diffusion in the determination of the mixing rate between the sprayed and unsprayed volumes.

As mentioned previously during the discussion of the analysis of the sump pH, the licensee applied an ultimate DF of 60, instead of a value of 100 as used in the CLB, to account for the determination that the sump pH will decrease below 7.0 after 24 hours.

Using the guidance from SRP 6.5.2, “Containment Spray as a Fission Product Cleanup System,” the licensee determined that the aerosol removal rate from the effects of the containment spray system, which actuates 2.34 minutes after the LOCA, is 6.9 per hour until a DF of 50 is reached at 2.185 hours post-LOCA. After the DF of 50 is reached the aerosol removal rate drops to 0.7 in accordance with the applicable regulatory guidance. In the CLB analysis the containment spray is continued until the aerosol DF reaches 1000 at 6.335 hours post-LOCA. In the AST analysis the aerosol removal rate of 0.7 per hour continues for the duration of the accident evaluation period. Section 3.3 of Appendix A of RG 1.183 states, "[t]here is no specified maximum DF for aerosol removal by sprays.” Therefore, in the AST analysis the licensee did not limit the DF for aerosol removal. In the licensee’s analysis, sprays are assumed to run indefinitely. The licensee asserts, and the staff agrees, that stopping the credit for sprays at 7.6 hours, when the aerosol DF reaches 1000, would have a very small impact on the calculated doses.

Using the guidance from SRP 6.5.2, the licensee determined that the elemental iodine removal rate from the effects of the containment spray system, which actuates 2.34 minutes after the LOCA, is in excess of 20 per hour. However, in accordance with the guidance in SRP 6.5.2, the licensee limited the removal rate constant for elemental iodine to 20 per hour. During the period of spray operation the elemental removal rate constant of 20 per hour from sprays was added to the elemental removal rate constant from wall deposition of 4.5 to yield an effective elemental removal rate constant of 24.5 per hour. The licensee applied this effective removal rate in the dose analysis from the time of spray actuation until 1.185 hours post-LOCA, at which time the DF for elemental iodine removal is calculated to be 60. As discussed previously, the licensee has determined that due to the consideration of the potential for iodine re-evolution resulting from the calculated sump pH dropping below 7, the ultimate elemental iodine DF is limited to 60.

This is a significant conservatism since by regulatory guidance the DF for elemental iodine is limited to 200.

The staff has reviewed the licensee's application of credit for iodine removal from the operation of the containment spray system and found that the analysis follows the applicable regulatory guidance, is conservative and is, therefore, acceptable.

3.1.1.3 Assumptions on Engineered Safety Feature (ESF) System Leakage

To evaluate the radiological consequences of ESF leakage, the licensee used the deterministic approach as prescribed in RG 1.183. This approach assumes that except for the noble gases, all of the fission products released from the fuel mix instantaneously and homogeneously in the containment sump water. Except for iodine, all of the radioactive materials in the containment sump are assumed to be in aerosol form and retained in the liquid phase. As a result, the licensee assumed that the fission product inventory available for release from ECCS leakage consists of 40 percent of the core inventory of iodine. This amount is the combination of 5 percent released to the containment sump water during the gap release phase and 35 percent released to the containment sump water during the early in-vessel release phase. This source term assumption is conservative in that 100 percent of the radioiodines released from the fuel are assumed to reside in both the containment atmosphere and in the containment sump concurrently. ECCS leakage develops when ESF systems circulate containment sump water outside containment and leaks develop through packing glands, pump shaft seals and flanged connections.

For the LOCA analysis of ESF leakage, the licensee used a value of 8280 cubic centimeters per hour (cc/hour), representing two times the maximum permitted recirculation loop leakage of 4140 cc/hour, as specified in RG 1.183, Appendix A, Item 5.2. As stated above, actual ECCS leakage would not begin until after the recirculation phase of the accident begins. For conservatism the licensee assumed that ESF leakage will start at the beginning of the LOCA and continue for the 30-day duration of the accident evaluation.

3.1.1.3.1 Assumptions on ESF System Leakage to the Fuel Handling Building (FHB)

RG 1.183, Appendix A, Regulatory Position 5.5, states that, "[i]f the temperature of the leakage is less than 212 °F or the calculated flash fraction is less than 10 percent, the amount of iodine that becomes airborne should be assumed to be 10 percent of the total iodine activity in the leaked fluid."

The licensee has determined that the pH of the containment sump will fall below 7.0 after 1 day. Therefore, the licensee considered that the fractional iodine release for ESF leakage would be greater than 10 percent as prescribed in RG 1.183. The licensee assumed that 16 percent of the iodine in the ESF leakage is released when the pH drops to 6.9, from 24 hours to 480 hours. The licensee assumed that 25 percent of the iodine in the ESF leakage is released when pH drops to 6.8, from 480 hours to 720 hours post-LOCA.

The licensee assumed that the ECCS leakage is released directly into the FHB and released instantaneously into the environment without credit for FHB Exhaust Air System filtration. The

licensee did not credit a reduction of activity released to the FHB as a result of dilution or holdup.

In accordance with RG 1.183, for ESF leakage into the FHB, the licensee assumed that the chemical form of the released iodine is 97 percent elemental and 3 percent organic.

3.1.1.3.1 Assumptions on ESF System Back-Leakage to the RWST

The licensee performed a calculation to determine the impact of a substantially degraded leakage condition with leak rates assumed to be approximately 10 times greater than the design leak rate for the ECCS isolation valves, thus allowing a greater than design leakage to migrate back to the RWST. The leakage values used in the analysis ranged from 180 to 480 cc/hour. The licensee considered the RWST suction line isolation valves, the low-head/high-head safety

injection pump's recirculation line isolation valves, and the containment spray pump's test line isolation valves for the three safety trains that are considered in this analysis. The licensee's analysis concluded:

- The motive force for leakage in the containment sump suction line is the high pressure in the containment resulting from the large break LOCA. The licensee determined that within the first 11.4 hours of an accident, the containment pressure will be reduced below the level needed to force contaminated water back into the RWST. Therefore, the licensee has concluded that no contaminated sump water will reach the RWST via this leak path.

- The containment spray pumps may be secured up to 13.4 days after initiation of a DBA LOCA and containment water will not reach the RWST via this leak path.

- The minimum time for leakage from valves assumed to have the degraded leak rate to reach the RWST following the initiation or the recirculation phase of the DBA LOCA is 44.1 days. This is beyond the LOCA dose analysis time period. At this point in time, the licensee determined that leakage into the RWST would be 1200 cc/hour.

The licensee provided additional information concerning the assessment of the potential for an ECCS leakage path back to the RWST. The licensee used the design-basis maximum seat leakage rate for each ECCS isolation valve to determine the rate of sump water back leakage into the RWST. The licensee then conservatively increased the maximum leakage rate by a factor of ten to determine the transport time for the leakage from the valve to the RWST, displacing all of the water in the piping. The analysis included the assumption that one valve in a series fails to close. The driving force for the leakage through the RWST suction line isolation valves is the post-accident pressure in the reactor containment building (RCB). The licensee assumed that all three trains of Containment Spray (CS), High-Head Safety Injection (HHSI), and Low-Head Safety Injection (LHSI) operate at accident initiation.

The licensee determined that a pressure differential of 22 pounds per square inch absolute (psia) would be required to transport water from the RCB sump to the RWST to overcome the elevation difference of 16.75 feet. The licensee determined that the pressure head on the RCB sump at recirculation switchover varies from 43 psia at switchover, to a maximum of 45 psia. The licensee determined that after approximately 11.4 hours, the RCB pressure would be less than the 22 psia required to transport water from the RCB sump to the RWST. The licensee determined that during this time period the distance the sump water would travel past the isolation valve farthest down stream would be approximately 0.15 feet, based on a flow rate of 8 cc/minute through a 16-inch line. Therefore, the licensee asserts, and the staff agrees, that the potential ECCS leakage from the RWST suction line isolation valves to the RWST will not impact the LOCA analysis results.

To evaluate the potential for leakage from the CS pumps to the RWST, the licensee assumed that the CS pumps would be secured after approximately 6.3 hours. During this time period, using a leakage rate of 3 cc/minute in a 6-inch pipe, the licensee determined that the contaminated sump water would only progress a distance of 0.21 feet. Using this methodology,

the licensee determined that it would take approximately 13.5 days for the sump water to reach the RWST via this pathway. The licensee stated that while emergency operating procedures would allow for CS pump operation for longer than 6.3 hours, the CS would be terminated much sooner than 13.5 days. Therefore, the licensee concludes, and the staff agrees, that the assumption that sump water will not reach the RWST via CS pump back-leakage is justifiable.

The licensee used a similar approach to determine that the shortest time for radioactive fluid to reach the RWST from safety-injection (SI) pump operation would be 44 days which is beyond the period of the LOCA dose analysis. Therefore, the licensee asserts, and the staff agrees, that the assumption that sump water will not reach the RWST via SI pump back-leakage is justifiable.

3.1.1.4 Assumptions on Containment Purging

The licensee evaluated the radiological effects of containment leakage via open supplemental purge lines which is assumed to occur for the first 23 seconds of the DBA LOCA. The release consists of RCS blowdown into the containment. The assumed volumetric flow rate from the supplemental purge lines is 142,000 cfm and is released to the environment via the plant vent.

During this time period of 23 seconds following accident onset, the licensee assumes that fuel failure has not occurred. This assumption follows the guidance in Table 4 of RG 1.183 which indicates that the initial release of the RCS into containment for a PWR would occur within the first 30 seconds of the accident prior to the onset of fuel damage. Per RG 1.183, the purge release evaluation should assume that 100 percent of the radionuclide inventory in the RCS liquid is released to the containment at the initiation of the LOCA and that this inventory should be based on the TS RCS equilibrium activity. The licensee based the RCS radionuclide concentrations for the AST analysis on 1 percent failed fuel (FF) which is also referred to as a clad defect. The 1 percent FF assumption for the AST analysis is a more conservative assumption than assuming that the RCS is at the TS equilibrium limit. Therefore, this conservative approach for the evaluation of the dose consequence of the RCS blowdown is acceptable to the staff.

The 23-second time period includes the signal and sequencer delays, Standby Diesel Generator startup time, and the valve closing time. This time does not include the 1.2 seconds between the postulated instantaneous break and the containment pressure reaching the setpoint value. However, the constant value of 142,000 cfm used for the choke flow through the ventilation system bounds the effect of neglecting these 1.2 seconds. During normal power operation, the Containment Supplementary Purge System vents the containment at 4500 cfm. However, for this analysis, the licensee used the maximum flow rate due to the pressure spike inside the containment.

The licensee used conservative assumptions to evaluate the containment purge contribution to the LOCA dose, and, therefore, the staff finds this evaluation acceptable for the AST LOCA analysis.

3.1.1.5 Control Room Habitability

The CRE is located at elevation 35 feet and in two heating, ventilating, and air conditioning (HVAC) rooms at elevations 10 feet and 60 feet in the Electrical Auxiliary Building as shown in Figure 6.4-1 of the Updated Final Safety Analysis Report (UFSAR).

The CR HVAC system is designed to maintain the CRE at a minimum of 0.125-inch water gauge positive pressure relative to the surrounding area, following postulated accidents with the exception of hazardous chemical/smoke releases. The CR HVAC system is activated on an ESF signal and/or high radiation in the outside air. The system is designed to introduce makeup air equivalent to the expected exfiltration air during plant emergency conditions. The design outside makeup airflow is 2000 cfm and is drawn from a single intake on the east side of the Electrical Auxiliary Building at elevation 80 feet 0 inches.

Additionally, during postulated accident conditions, on detection of high radiation in the outside air or on an SI signal, outside makeup air for the CRE is automatically routed through makeup air units and cleanup units containing charcoal filters. Additionally, during postulated accident conditions, CR air is routed through recirculation air cleanup units containing charcoal filters at a flow rate of 10,000 cfm.

3.1.1.5.1 CR Ventilation Assumptions

The licensee's assumption of 100 cfm unfiltered in-leakage is validated by in-leakage testing conducted on Unit 1 during March 2004 and on Unit 2 during March 2007. The testing was conducted using the tracer gas method described in American Society for Testing and Materials (ASTM) E741-00, "Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution." The test results for Unit 1 were reported in a letter dated August 3, 2004 (ADAMS Accession No. ML042260183), in response to NRC Generic Letter 2003-01, "Habitability." The limiting train combination test results were 9.4 +/- 50 standard cubic feet per minute (scfm) in Unit 1 and 64 +/- 8 scfm in Unit 2. Therefore, the staff concludes that an unfiltered in-leakage assumption of 100 cfm conservatively bounds the test results.

In contrast to the CLB, the revised AST analyses assume that in the emergency mode all makeup flow is unfiltered. The total of 4 inches of charcoal filtration per train which consists of 2 inches of the makeup filters and 2 inches of the cleanup filters, for make-up air in the CLB analysis is not credited in the AST analysis. Only the recirculation filtration is credited with 95 percent removal of both elemental and organic iodine and 99 percent removal of particulates. Therefore, the assumed filtered make-up air flow is 0 cfm and the 2200 cfm make-up flow is added to the 100 cfm unfiltered in-leakage value. The 100 cfm unfiltered in-leakage values includes the contribution from door pumping action from CR ingress and egress. The AST analysis assumes that a total of 2300 cfm directly enters the CR without filtration. In addition, the licensee does not credit the use of non-ESF ventilation systems during the DBA analysis. The CR parameters used in the AST analyses are shown in Table 4 of this SE.

3.1.1.5.2 Direct Shine Dose Evaluations

The total CR LOCA dose includes direct shine contributions from the following DBA-LOCA radiation sources:

- Contamination of the CR atmosphere by the intake and infiltration of the radioactive material contained in the radioactive plume released from the facility,
- Direct shine from the external radioactive plume released from the facility with credit for CR structural shielding,
- Direct shine from radioactive material in the containment with credit for both the containment and CR structural shielding,
- Direct shine from radioactive material leaked from the containment electrical penetrations into the electrical penetration area, with credit for CR structural shielding, and
- Radiation shine from radioactive material in systems and components inside or external to the CR envelope including radioactive material buildup on the CR ventilation filters.

RG 1.196 defines the CRE as follows: "[t]he plant area, defined in the facility licensing basis, that in the event of an emergency, can be isolated from the plant areas and the environment external to the CRE. This area is served by an emergency ventilation system, with the intent of maintaining the habitability of the CR. This area encompasses the CR, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident."

The licensee evaluated the DBA LOCA radiation dose to personnel in the CR from the following sources: gamma shine from the primary containment airborne activity, airborne activity in the electrical penetration area, activity in the radioactive cloud surrounding the plant structures, and from trapped activity on filters. The evaluation of the contribution of the shine from the electrical penetration area represents an additional source that was not evaluated in the CLB. The results of the licensee's evaluations of the CR gamma shine dose are included in Table 4 of this SE.

The licensee has determined that the time-integrated activity of radioiodine airborne in the containment would be approximately an order of magnitude lower for the AST than for the STP CLB which is based on the TID-14844 source term. The licensee asserts, and the staff agrees, that the TID-14844 source term will produce larger gamma energies and larger integrated gamma doses in the containment atmosphere than the AST. This conclusion is based on the several considerations including the fact that the TID-14844 source term is assumed to be instantaneous and contains a larger release of iodine than the time-dependent AST. In addition, the containment removal mechanisms are more effective for the AST due to the assumption of an increased percentage of the iodine release being in aerosol form. Even though the AST includes the airborne release of significant quantities of additional non-iodine

activity in particulate form, this activity is readily removed by the containment spray system. As a result, the licensee has concluded, and the staff agrees, that the gamma shine contribution to CR dose from the containment will be less for the AST than for the STP CLB TID-14844 source term.

The licensee has determined that the STP CR CLB shine dose due to activity airborne in the containment of 0.101 rem is bounding for the AST evaluation. Therefore, the licensee used a value of 0.101 rem for the contribution due to containment shine for the AST CR dose consequence analysis. In a similar fashion, the licensee has determined that the CLB shine dose to the TSC due to the RCB airborne activity of 0.004 rem is bounding for the AST analysis.

The licensee considered the gamma shine component to CR dose from activity in the Electrical Penetration area that is directly between the CRE and the containment building. This source was not considered in the CLB CR LOCA dose analysis. The licensee evaluated this component to CR shine by incorporating an additional compartment to the RADTRAD plant model. The licensee used the maximum post-LOCA containment temperature and pressure to convert the electrical penetration tested mass leak rate to a volumetric leak rate for use in the dose analysis model. The airborne activity as a function of time within this compartment was calculated and a dose calculation was performed for shine dose in the CR using the MicroShield code. MicroShield is a point kernel integration code used for general-purpose gamma shielding analysis and is appropriate for use in this application. The licensee determined the dose contribution from this source to be 0.0174 rem.

To evaluate the shine from the radioactive cloud outside the CR, the licensee adjusted the shine dose increment at the outer boundary of the LPZ by the ratio of the maximum onsite atmospheric dispersion factor (χ/Q) value to that for the LPZ for each χ/Q averaging period. The results were reduced by using shielding attenuation factors for the CR and TSC and the final shine dose was obtained by adding the increments for each averaging period. The dose contribution from this source is 0.014 rem to the CR and 0.212 rem to the TSC.

The licensee provided the following additional information describing the basis for the shielding factors used in the analysis of direct dose to the CR and to the TSC. The CR shielding attenuation factor of 1.03E-3 and the TSC shielding attenuation factor of 1.56E-2 were taken from the CLB calculations as the factors afforded by 2.5 feet of concrete shielding and 1.5 feet of concrete shielding, respectively, for sources outside the CR and TSC. The licensee asserts, and the staff agrees, since cloud shine doses are dominated by noble gas releases, and since noble gas releases are similar for both the CLB and the AST, the use of the CLB shielding attenuation factors are appropriate for use in the AST evaluation.

To evaluate the CR gamma shine from trapped activity on filters, the licensee used a similar approach to the treatment of the shine from the containment by making a comparison of source gamma energies as a function of time to either justify the use of the CLB calculated dose or to make adjustments to the CLB value for use in the AST evaluation.

As discussed earlier, the licensee determined that the time integrated activity of radioiodine airborne in the containment would be substantially less for the AST than for the STP CLB. The airborne noble gas activity is comparable for both the AST and the STP CLB TID-14844 source

term. While the AST involves the airborne release of significant quantities of additional non-iodine activity, because it is in particulate form, this activity is readily removed by the containment spray system and natural plate out. In addition, the external gamma dose contribution from non-iodine airborne particulates assumed to be released for the AST is substantially less than that for the iodine isotopes. For these reasons, the licensee has concluded, and the staff agrees, that radiation from the activity trapped on filters will be less for the AST than for the STP CLB.

The licensee's CLB for both the CR and the TSC includes filter shine contributions. The CLB evaluation has determined that the only significant filter shine contributions are the CR shine dose of 0.00218 rem due to the CR recirculation make-up filters, and the TSC shine dose of 0.844 rem due to the CR make-up filters.

The AST analysis for CR inhalation and immersion doses does not credit the CR make-up filters. Therefore, to conservatively bound the gamma shine dose from the CR make-up filter loading, the licensee performed a separate calculation assuming a filter efficiency of 100 percent.

To assess the dose contribution from the CR recirculation filter loading, the licensee conservatively did not credit the iodine removal from the make-up filters which resulted in a slightly higher loading late in time for the AST analysis. This conservative approach resulted in an increase in the CR dose from the CR clean-up filters over the value used in the CLB evaluation.

Based on the above, the NRC staff finds that the potential direct shine dose contributions to the CR LOCA dose analysis used conservative assumptions and sound engineering judgment and is, therefore, acceptable.

The licensee evaluated the radiological consequences resulting from the postulated LOCA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose reference values provided in 10 CFR 50.67 and the accident specific dose guidelines specified in SRP Section 15.0.1. The staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the staff are presented in Table 5 and the licensee's calculated dose results are given in Table 1. The staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the LOCA meet the applicable accident dose criteria and are, therefore, acceptable.

3.1.2 Fuel Handling Accident (FHA)

The FHA involves the drop of a fuel assembly on top of other fuel assemblies during refueling operations. The mechanical part of the licensee's analysis remains unchanged from the CLB and assumes that the total number of failed fuel rods is 314 out of the 50,952 rods in an entire core. The depth of water over the damaged fuel is not less than 23 feet and is controlled by TS 3/4.9.10 and 3/4.9.11. Following reactor shutdown, decay of short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. The proposed AST

amendment takes credit for the normal decay of irradiated fuel rather than crediting certain active ventilation filtration systems credited in the CLB FHA.

The licensee's analysis is fully compliant with RG 1.183. The analysis was performed assuming a decay period of 42 hours after shutdown and assuming a ground-level release through the plant vent. An FHA in either the containment or the FHB would involve a release via the plant vent or possibly directly from the containment. However, a release directly from the containment would experience more favorable atmospheric dispersion on the path to the CR and TSC air intake than a release from the plant vent because of the greater distance involved. Therefore, the licensee used the plant vent-release pathway for all FHA scenarios to ensure conservative results.

3.1.2.1 Source Term

The fission product inventory that constitutes the source term for this event is the gap activity in the fuel rods assumed to be damaged as a result of the postulated design-basis FHA. Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod cladding during normal power operations. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released to the surrounding water as a result of the accident. The licensee performed a detailed analysis to ensure that the most restrictive case would be considered for the FHA dose consequence analysis.

Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the reactor cavity or SFP depending on their physical and chemical form. Following the guidance in RG 1.183, Appendix B, Regulatory Position 1.3, the licensee assumed: (1) that the chemical form of radioiodine released from the fuel to the spent fuel pool consists of 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide, (2) the CsI released from the fuel is assumed to completely dissociate in the pool water, and (3) because of the low pH of the pool water, the CsI re-evolves and releases elemental iodine. This results in a final iodine distribution of 99.85 percent elemental iodine and 0.15 percent organic iodine. The licensee assumed that the release to the pool water and the chemical redistribution of the iodine species occurs instantaneously.

As corrected by item 8 of Regulatory Issue Summary (RIS) 2006-04 (ADAMS Accession No. ML053460347), RG 1.183, Appendix B, Regulatory Position 2, should read as follows, "[i]f the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 285 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5 percent of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85 percent) and organic iodine (0.15 percent) species results in the iodine above the water being composed of 70 percent elemental and 30 percent organic species."

In accordance with RG 1.183, Appendix B, Regulatory Position 2, the licensee credited an overall iodine DF of 200 for a water cover depth of 23 feet. Consistent with RG 1.183, the licensee credited an infinite DF for the remaining particulate forms of the radionuclides

contained in the gap activity. In accordance with RG 1.183, the licensee did not credit decontamination from water scrubbing for the noble gas constituents of the gap activity.

The licensee used ORIGEN 2.1 to calculate plant-specific fission product inventories for use in the FHA dose analyses. The fraction of the core that is damaged is assumed to be one fuel assembly which consists of 264 fuel rods plus an additional 50 fuel rods in an impacted assembly. This results in a total of 314 fuel rods out of the 50,952 rods in the full core. A peaking factor of 1.7 was applied to the fission product inventory of the damaged rods. This peaking factor value is a practical bounding value for the peaking factors found in the cycle-specific Core Operating Limits Report (COLR), based on previous core design history and future projections.

The licensee analyzed the FHA based on the fuel rod gap activity release fractions from RG 1.183, Regulatory Position 3, Table 3. RG 1.183 states that these release fractions have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kW/ft peak rod average power for burnups exceeding 54 gigawatt days per metric ton of uranium (GWD/MTU).

The licensee performed an evaluation to determine the best estimate fuel rod average burnup and power for typical 18-month cycle designs. The licensee's evaluation indicates that the RG 1.183 limitation of 6.3 kW/ft peak rod average power for burnups exceeding 54,000 MWD/MTU for the application of the AST is met with significant margin. The licensee has determined for burnups exceeding 54,000 MWD/MTU, the highest corresponding linear heat rate is 5.7 kW/ft.

In accordance with RG 1.183, alkali metals are released as particulates and are assumed to experience an infinite DF due to the water submergence. Therefore, the licensee did not include the alkali metals cesium and rubidium in the FHA dose analysis.

3.1.2.2 Transport

Releases from the FHB are via the plant vent. Releases from the RCB purge and RCB Personnel Airlock are also vented to the atmosphere from the same plant vent. The atmospheric dispersion factors for a release from the plant vent to the CR intake are more limiting than for releases from the RCB equipment hatch opening since the plant vent is much closer to the CR air intake than the Equipment Hatch. Therefore, to evaluate the FHA releases, the licensee used the atmospheric dispersion factors from the plant vent to the CR intake to conservatively encompass all FHA scenarios.

Although RG 1.183 allows for the environmental release of activity from an FHA to occur over a 2-hour period, the licensee conservatively assumed an instantaneous release of the fission products to the environment with no credit for holdup or dilution in the surrounding structures.

3.1.2.3 CR Habitability for the Fuel Handling Accident

The licensee evaluated CR habitability for the FHA assuming that the activity is released directly to the CR HVAC intake from the plant vent using the plant vent to CR atmospheric dispersion factors. The licensee conservatively assumed that the CR internal air is in equilibrium with the air outside the CR HVAC intake. The licensee did not credit CR pressurization or any other safety functions of the CR HVAC systems in the FHA analysis.

The licensee evaluated the radiological consequences resulting from a postulated FHA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident-specific dose criteria specified in SRP Section 15.0.1. The staff's review has found that the licensee used analysis assumptions and input consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the staff are presented in Table 6 and the licensee's calculated dose results are given in Table 1. The staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the FHA meet the applicable accident dose criteria and are, therefore, acceptable.

3.1.3 Main Steam Line Break (MSLB) Accident

The postulated MSLB accident assumes a double-ended break of one main steam line outside the primary containment. This leads to an uncontrolled release of steam from the steam system. The resultant depressurization of the steam system causes the MSIVs to close and, if the plant is operating at power when the event is initiated, causes the reactor scramm. For the MSLB DBA radiological consequence analysis, a LOOP is assumed to occur shortly after the trip signal. Following a reactor trip and turbine trip, the radioactivity is released to the environment through the steam generator (SG) power-operated relief valves (PORVs). Because the LOOP renders the main condenser unavailable, the plant is cooled down by releasing steam to the environment.

The radiological consequences of an MSLB outside containment will bound the consequences of a break inside containment. Therefore, only the MSLB outside of containment is considered with regard to the radiological consequences. The affected SG, hereafter referred to as the faulted SG, rapidly depressurizes and releases the initial contents of the SG to the environment. The MSLB accident is described in Section 15.1.5 of the STP UFSAR. RG 1.183, Appendix E, identifies acceptable radiological analysis assumptions for a PWR MSLB.

As stated above, the steam release from a rupture of a main steam line would result in an initial increase in steam flow, which decreases during the accident as the steam pressure decreases. The increased energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive rod cluster control assembly (RCCA) is assumed stuck in its fully withdrawn position after the reactor trip, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid delivered by the SI system.

3.1.3.1 Source Term

Appendix E of RG 1.183 identifies acceptable radiological analysis assumptions for a PWR MSLB accident. RG 1.183, Appendix E, Regulatory Position 2, states that if no or minimal fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity allowed by TS including the effects of pre-accident and concurrent iodine spiking. The licensee's evaluation indicates that no fuel damage would occur as a result of an MSLB accident.

Therefore, the licensee considered the two radioiodine spiking cases described in RG 1.183. The first case is referred to as a pre-accident iodine spike and assumes that a reactor transient has occurred prior to the postulated MSLB that has raised the primary coolant iodine concentration to the maximum value permitted by the TS for a spiking condition. For STP, the maximum iodine concentration allowed by TS as the result of an iodine spike is 60 microcuries per gram (µCi/gm) DEI.

The second case assumes that the primary system transient associated with the MSLB causes an iodine spike in the primary system. This case is referred to as a concurrent iodine spike. The increase in primary coolant iodine concentration for the concurrent iodine spike case is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the TS limit for normal operation. For STP the RCS TS limit for normal operation is 1.0 µCi/gm DEI. The duration of the concurrent iodine spike is assumed to be 8 hours in accordance with the applicable guidance.

RG 1.183, Appendix E, Regulatory Position 4 states that, "[t]he chemical form of radioiodine released from the fuel should be assumed to be 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine releases from the SGs to the environment should be assumed to be 97 percent elemental and 3 percent organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking."

Since RG 1.183 specifies that the chemical form of particulate iodine is cesium iodide (CsI), the licensee assumed that the primary system transient that causes the iodine spiking also increases the cesium and rubidium concentrations in the RCS in relative amounts.

The licensee took a conservative exception to the Appendix E statement that the iodine release from the SG should be 97 percent elemental and 3 percent organic. The licensee asserts, and the staff agrees, that the noted speciation is based on the assumption that none of the particulate iodine, which makes up 95 percent of the RCS flow mixing into the bulk SG water, is released to the atmosphere.

RG 1.183, Appendix E, Regulatory Position 5.5.4 states that, "[t]he radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators."

The licensee asserts, and the staff agrees, that the guidance contains an apparent contradiction in that in Appendix E, Section 4, the particulate iodine activity is assumed to be retained in the SG water and not released to the atmosphere while Section 5.5.4 provides guidance on the treatment of particulate iodine activity released from the SGs.

To address this issue, the licensee applied the guidance in Appendix E, Section 4 to the release of iodine incorporating a partition coefficient (PC) of 100 for particulate and elemental iodine but not for organic iodine. When the PC of 100 is applied as stated to the chemical form of radioiodine released from the fuel, the resulting speciation for an SG release becomes 4.2 percent elemental iodine, 13.1 percent organic iodide and 82.7 percent particulate iodine. This speciation is a departure from the RG 1.183 guidance on the speciation for an SG release of 97 percent elemental iodine, 3 percent organic iodide with no particulate iodine released.

The licensee asserts, and the staff agrees, that this departure from regulatory guidance results in a larger release of iodine since particulates are released and no PC is used to reduce the amount of organic iodine released.

For the MSLB accident, the licensee evaluated the radiological dose contribution from the release of secondary-side activity using the equilibrium secondary-side specific activity TS LCO of 0.1 μCi/gm DEI.

3.1.3.2 Transport

With the exception of the iodine speciation for releases from the SGs, the licensee followed the guidance as described in RG 1.183, Appendix E, Regulatory Position 5 in all other aspects of the transport analysis for the MSLB.

For additional conservatism the licensee assumed a total primary-to-secondary leak rate equal to 1 gallons per minute (gpm) which is higher than the TS total allowable leak rate of 0.42 gpm. The licensee modeled the assumed 1 gpm primary-to-secondary leakage as 0.65 gpm for the three intact SGs and 0.35 gpm for the faulted SG.

RG 1.183, Appendix E, Regulatory Position 5.2, states that, "[t]he density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., [pounds of mass per hour] lbm/hour) should be consistent with the basis of the parameter being converted. The alternate repair criteria (ARC) leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate TSs are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc [grams per cubic centimeter] (62.4 [pounds of mass per cubic foot] lbm/ft^3)." The licensee's leak rate testing results are adjusted so that the allowable leakage corresponds to a density of 8.33 pounds per gallon and accordingly this density was used to convert the volumetric leak rate to a total mass flow rate due to SG tube leakage in the MSLB dose consequence analysis.

RG 1.183, Appendix E, Regulatory Position 5.3, states that, "[t]he primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 degrees Celsius (°C)

(212 °F). The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated." In accordance with RG 1.183 and existing licensing basis the licensee assumed that primary-to-secondary leakage continues for 8 hours after the MSLB at which time the residual heat removal (RHR) system is assumed to begin operation.

In accordance with RG 1.183, the licensee assumed that all noble gas radionuclides released from the primary system are released to the environment without reduction or mitigation. Following the guidance from RG 1.183, Appendix E Regulatory Positions 5.5.1, 5.5.2, and 5.5.3, the licensee assumed that all of the primary-to-secondary leakage into the faulted SG will flash to vapor, and be released to the environment with no mitigation. For the unaffected SGs that are used for plant cooldown, the licensee assumed that the primary-to-secondary leakage mixes with the secondary water without flashing.

RG 1.183, Appendix E, Regulatory Position 5.5.4, states that, "[t]he radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators."

Accordingly, the licensee assumed that the radioactivity in the bulk water of the unaffected SGs becomes vapor at a rate that is a function of the steaming rate and the partition coefficient. The licensee used a partition coefficient of 100 for elemental iodine and other particulate radionuclides released from the intact SGs. As previously discussed, the licensee did not credit the partitioning of organic iodine. The licensee assumed that the organic iodine migrates directly to the steam space and become immediately available for release.

The licensee credited operator action to isolate the faulted SG within 30 minutes of the event. The total release from the faulted SG is 214,000 lbm initially plus a subsequent release of 385,000 lbm from the Main Feedwater System and the Auxiliary Feedwater System, for a total release of 599,000 lbm.

Eight (8) hours after the accident, the RHR system is in operation and no further steam containing radionuclides is released from the SGs to the environment except for the assumption of the leakage through the MSIV above-seat orifices. The licensee assumed that releases from the orifices continue until 36 hours after the start of the accident. This is conservative since all releases would terminate in less than 8 hours when the RHR system is operational.

The licensee provided the following additional information to explain the basis for the conservative assumption that releases from the orifices continue until 36 hours after the start of the accident. The time used for the release through the MSIV above-seat orifices is taken from the TS requirement for plant cooldown to Mode 5 following an SGTR or MSLB event. Per TS Section 3.6.3, the plant is required to be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours. Since the release through the 3/8-inch orifices is very small compared to the total release, the licensee has taken a conservative approach by including the dose consequence due to this additional release path.

The release of 1.93 lbm/seconds or 1.86 cfm per loop is calculated assuming critical flow through the orifices at 1200 psia. The leakage is assumed to be constant for the entire 36-hour period, regardless of the impact to the secondary side of cooling the plant to the point that the RHR system is placed in operation. The steam release through the orifices is assumed to be released directly to the environment when in reality the actual flow path would be to the condenser.

The licensee assumed that the break and the above-seat drain releases occur in the Isolation Valve Cubicle next to the PORVs. Therefore, the PORV-to-CR χ/Qs are used for the CR and TSC dose analyses.

3.1.3.3 CR Ventilation Assumptions for the MSLB

The licensee evaluated CR habitability for the MSLB assuming that the CR ventilation system automatically transfers to the emergency mode of operation after the initiation of safety injection. The revised AST CR analyses assume that in the emergency mode all makeup flow is unfiltered. The total of 4 inches of charcoal filtration per train, which consists of 2 inches for the makeup filters and 2 inches of the cleanup filters, is not credited in the AST analysis. Only the recirculation filtration is credited with 95 percent removal of both elemental and organic iodine and 99 percent removal of particulates. Therefore, the assumed filtered make-up air flow is 0 cfm and the 2200 cfm make-up flow is added to the 100 cfm unfiltered in-leakage value. The 100 cfm unfiltered in-leakage values includes the contribution from door pumping action from CR ingress and egress. The AST analysis assumes that a total of 2300 cfm directly enters the CR without filtration. In addition, the licensee does not credit the use of non-ESF ventilation systems during the DBA analysis. The CR parameters used in the AST analyses are shown in Table 4 of this SE.

The licensee evaluated the radiological consequences resulting from the postulated MSLB accident and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose reference values provided in 10 CFR 50.67 and accident-specific dose criteria specified in SRP Section 15.0.1. The staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the staff are presented in Table 7 and the licensee's calculated dose results are given in Table 1. The staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the MSLB meet the applicable accident dose guidelines and criteria, and, therefore, licensee's analysis is acceptable.

3.1.4 Steam Generator Tube Rupture (SGTR) Accident

The licensee evaluated the radiological consequences of an SGTR accident as a part of the full implementation of an AST. The SGTR accident is evaluated based on the assumption of an instantaneous and complete severance of a single SG tube. The postulated break allows primary coolant liquid to leak to the secondary side of the ruptured SG. Integrity of the barrier between the RCS and the main steam system is significant from a radiological release standpoint. The radioactivity from the leaking SG tube mixes with the shell-side water in the

affected SG. For the SGTR DBA radiological consequence analysis, a LOOP is assumed to occur shortly after the trip signal. Following a reactor trip and turbine trip, the radioactive fluid is released to the environment through the SG PORVs. Because the LOOP renders the main condenser unavailable, the plant is cooled down by releasing steam to the environment.

In the STP SGTR analysis, operators are assumed to identify the ruptured SG and attempt to close the PORV on the ruptured SG within 10 minutes. Subsequently the PORV is assumed to fail open which is the single failure for this accident scenario. The analysis then credits operator action to isolate the failed open PORV by manually closing the PORV block valve within 15 minutes of the PORV failure. Therefore, the steam release via the ruptured SG's PORV is assumed to continue for a total of 25 minutes. These assumptions are consistent with the CLB SGTR analysis.

The Westinghouse NSAL-06-15, dated December 13, 2006, advised operators of Westinghouse plants that the single-failure scenario for the SGTR analysis may not be limiting. The methodology included evaluations of various single failures from a reference plant. Recent industry operating experience identified a condition where a failed-open MSIV on the steamline from the ruptured SG may result in a steam flow that is higher than that previously assumed in the accident analysis and thus higher offsite dose consequences.

The STP current SGTR analysis and the SGTR analysis presented in the SE for this LAR assumes a failed open SG PORV as the limiting single failure as far as assumed total steam release. The licensee's evaluation of NSAL-06-15 has resulted in a revised conclusion that the failed open MSIV results in a greater steam release at STP. This is because the steam valves to the moisture-separator reheater fail open on a loss of instrument air resulting from a loss of offsite power. The steam valves to the moisture-separator reheater fail closed for the reference plant thus significantly reducing the steam release from a failed open MSIV.

As a result of this finding, the licensee is currently operating under an administrative limit for RCS DEI that is lower than the TS limit. STP will continue to maintain an administrative limit for RCS DEI until a permanent solution is implemented so that the radiological dose guidelines for the SGTR analysis remain bounding, and the licensee will continue to comply with GDC 19.

The licensee provided additional information describing the potential CR and offsite dose consequences for an SGTR accident assuming that the RCS specific activity was at the TS 3.4.8, "RCS Specific Activity" LCO and with a failed open MSIV as the limiting single-failure assumption. The evaluation showed that the offsite dose consequences would meet the applicable offsite dose guidelines. The analysis demonstrated to the satisfaction of the staff that, with the administrative limit for RCS DEI, the CR dose would be reduced to a value well within the applicable limits of GDC 19. At all times, the licensee must comply with the GDC 19 CR dose limit in 10 CFR 50.67(b)(2)(iii) and, therefore, the staff finds that the licensee's approach to maintain RCS DEI at an appropriate administrative limit is an acceptable solution until a permanent solution is implemented.

3.1.4.1 Source Term

Appendix F of RG 1.183 identifies acceptable radiological analysis assumptions for an SGTR accident. If a licensee demonstrates that no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by TS. Two radioiodine spiking cases are considered. The first case is referred to as a pre-accident iodine spike and assumes that a reactor transient has occurred prior to the postulated SGTR that has raised the primary coolant iodine concentration to the maximum value permitted by the TS for a spiking condition. For STP, the maximum iodine concentration allowed by TS as a result of an iodine spike is 60 µCi/gm DEI.

The second case assumes that the primary system transient associated with the SGTR causes an iodine spike in the primary system. This case is referred to as a concurrent iodine spike. Initially, the plant is assumed to be operating with the RCS iodine activity at the TS limit for normal operation. For STP, the RCS TS limit for normal operation is 1 µCi/gm DEI. The increase in primary coolant iodine concentration for the concurrent iodine spike case is estimated using a spiking model that assumes that as a result of the accident, iodine is released from the fuel rods to the primary coolant at a rate that is 335 times greater than the iodine equilibrium release rate corresponding to the iodine concentration at the TS limit for normal operation. The iodine release rate at equilibrium is equal to the rate at which iodine is lost due to radioactive decay, RCS purification, and RCS leakage. The iodine release rate is also referred to as the iodine appearance rate. The concurrent iodine spike is assumed to persist for a period of 8 hours.

The licensee's evaluation indicates that no fuel damage is predicted as a result of an SGTR accident. Therefore, consistent with the CLB and regulatory guidance, the licensee performed the SGTR accident analyses for the pre-accident iodine spike case and the concurrent accident iodine spike case. In accordance with regulatory guidance, the licensee assumed that the activity released from the iodine spiking mixes instantaneously and homogeneously throughout the primary coolant system.

Since RG 1.183 specifies that the chemical form of particulate iodine is cesium iodide (CsI), the licensee assumed that the primary system transient that causes the iodine spiking also increases the cesium and rubidium concentrations in the RCS in relative amounts.

As discussed for the MSLB, the licensee applied the guidance in RG 1.183, Appendix E, Section 4, to the release of iodine from the SGs incorporating a PC of 100 for particulate and elemental iodine but not for organic iodine. When the PC of 100 is applied as stated to the chemical form of radioiodine released from the fuel, the resulting speciation for an SG release becomes 4.2 percent elemental iodine, 13.1 percent organic iodide and 82.7 percent particulate iodine. As stated previously this speciation is a conservative departure from the RG 1.183 guidance on the speciation for an SG release of 97 percent elemental iodine, 3 percent organic iodide with no particulate iodine released. For the SGTR the licensee included the radiological dose contribution from the release of secondary coolant iodine activity at the TS limit of 0.1 µCi/gm DEI.

3.1.4.2 Transport

With the exception of the conservative treatment of the iodine speciation for releases from the SGs, the licensee followed the guidance as described in RG 1.183, Appendix F, Regulatory Position 5 in all other aspects of the transport analysis for the SGTR.

For conservatism the licensee assumed a total primary-to-secondary leak rate equal to 1 gpm which is higher than the TS total allowable leak rate of 0.42 gpm. The licensee modeled the assumed 1 gpm primary-to-secondary leakage as 0.65 gpm for the three intact SGs and 0.35 gpm for the ruptured SG.

RG 1.183, Appendix E, Regulatory Position 5.2, states that, "[t]he density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hour) should be consistent with the basis of the parameter being converted. The alternate repair criteria (ARC) leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate TSs are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/square feet)." The licensee's leak rate testing results are adjusted so that the allowable leakage corresponds to a density of 8.33 pounds per gallon and accordingly this density was used to convert the volumetric leak rate to a total mass flow rate due to SG tube leakage in the SGTR dose consequence analysis.

RG 1.183, Appendix F, Regulatory Position 5.3, states that, "[t]he primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 °C (212 °F). The release of radioactivity from the unaffected SGs should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated." The licensee assumed that the release of radioactivity from both the ruptured SG and the unaffected SGs continues for 8 hours, until the RHR system is in operation, and steam releases from the SGs have been terminated.

The licensee evaluated the dose consequences from discharges of steam from the intact SGs for a period of 8 hours, until the primary system has cooled sufficiently to allow an alignment to the RHR system. At this point in the accident sequence, steaming is no longer required for cool down and releases from the intact SG are terminated. The licensee conservatively increased the steam flow rates calculated from the thermal hydraulic model by a factor of 1.4 to add additional margin to the SGTR dose consequence analysis.

The licensee assumed that the source term resulting from the radionuclides in the primary system coolant, including the contribution from iodine spiking, is transported to the ruptured SG by the break flow. A portion of the break flow is assumed to flash to steam because of the higher enthalpy in the RCS relative to the secondary system. The licensee assumed that the flashed portion of the break flow will ascend through bulk water of the SG, enter the steam space of the affected generator, and be immediately available for release to the environment with no credit taken for scrubbing. Although RG 1.183 allows the use of the methodologies described in NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," May 1985, to determine the amount of scrubbing

credit applied to the flashed portion of the break flow, the licensee did not credit scrubbing of the activity in the break flow in the ruptured SG.

During the first 65.5 seconds of the event, prior to the reactor trip and the assumed concurrent LOOP, the licensee assumed that all of the SG flow is routed to the condenser. The licensee applied an additional DF of 100, to the condenser releases. Therefore, the steam released from the condenser during the first 66.5 seconds following the SGTR has a total iodine DF of 10,000 which includes a DF of 100 as a result of changing phase in the SG and an additional DF of 100 exiting through the condenser. After 66.5 seconds, the condenser is no longer available due to the assumed LOOP. Therefore, the additional condenser DF of 100 is only applied to the flashed flow for the first 65.5 seconds of the event.

The iodine and other non-noble gas isotopes in the non-flashed portion of the break flow are assumed to mix uniformly with the SG liquid mass and be released to the environment in direct proportion to the steaming rate and in inverse proportion to the applicable partition coefficient (PC).

The licensee has determined that for the SGTR accident, both SGs effectively maintain tube coverage. In accordance with RG 1.183, Appendix E, Regulatory Position 5.5.1, the licensee assumed that for the ruptured SG, and the unaffected SGs used for plant cooldown, the primary-to-secondary leakage mixes with the secondary water without flashing due to the total submergence of the SG tubes. The iodine and other non-noble gas isotopes in the primary-to-secondary leakage flow are assumed to mix uniformly with the SG liquid mass and be released to the environment in direct proportion to the steaming rate and in inverse proportion to the applicable PC.

As discussed previously, the licensee's evaluation of the releases from the steaming of the liquid mass in the SGs credits a PC of 100 for all non-noble gas isotopes with the exception of organic iodine. Following the applicable regulatory guidance, the licensee assumed that all noble gas radionuclides released from the primary system are released to the environment without reduction or mitigation.

3.1.4.3 <u>CR Ventilation Assumptions for the SGTR</u>

The licensee evaluated CR habitability for the SGTR assuming that the CR ventilation system automatically transfers to the emergency mode of operation after the initiation of safety injection. This is assumed to happen concurrently with the SGTR instead of at the time of the reactor trip which is assumed to be 66.5 seconds after the SGTR event. As previously discussed the licensee built a considerable margin of safety in the SGTR dose analysis by increasing the mass released by a factor of 1.4 over the amounts predicted to be released by the thermal/hydraulic modeling. The licensee asserts, and the staff agrees, that since the mass releases are increased by about 40 percent, the 66.5 second time difference is negligible. In addition, since the LOOP is assumed to occur concurrently with the reactor trip, SGTR releases before the LOOP are assumed to be routed to the condenser where an additional DF of 100 significantly reduces the radiological consequences.

The revised AST CR analyses assume that in the emergency mode all makeup flow is unfiltered. The total of 4 inches of charcoal filtration per train, which consists of 2 inches for the makeup filters and 2 inches of the cleanup filters, is not credited in the AST analysis. Only the recirculation filtration is credited with 95 percent removal of both elemental and organic iodine and 99 percent removal of particulates. Therefore, the assumed filtered make-up air flow is 0 cfm and the 2200 cfm make-up flow is added to the 100 cfm unfiltered in-leakage value. The 100 cfm unfiltered in-leakage values includes the contribution from door pumping action from CR ingress and egress. The AST analysis assumes that a total of 2300 cfm directly enters the CR without filtration. In addition, the licensee does not credit the use of non-ESF ventilation systems during the DBA analysis. The CR parameters used in the AST analyses are shown in Table 4 of this SE.

The licensee evaluated the radiological consequences resulting from the postulated SGTR accident and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident dose criteria specified in SRP Section 15.0.1. The staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the staff are presented in Table 8 and the licensee's calculated dose results are given in Table 1. The staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the SGTR accident meet the applicable accident dose criteria and are, therefore, acceptable.

3.1.5 Control Rod Ejection Accident (CREA)

Section 15.4.8 of the STP UFSAR describes the CREA as the mechanical failure of a control rod mechanical pressure housing such that the coolant system pressure ejects an RCCA and drive shaft to a fully withdrawn position. The consequences of this mechanical failure are a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. Following the applicable guidance, the licensee evaluated two separate release scenarios for the CREA. In the first case, the CRE is assumed to induce a LOCA resulting in a release of fission products into the containment atmosphere and a subsequent release to the environment from the containment leakage pathway.

For the second case, the radiological consequences from a CRE are evaluated assuming that the RCS boundary remains intact and that fission products are released to the environment from the secondary system. In this case, fission products from the damaged fuel are assumed to be released to the primary coolant and transported to the secondary system through primary-to-secondary leakage in the SGs. The CREA is analyzed with the assumption of a concurrent LOOP which causes steam releases from the secondary system to occur through the SG PORVs and safety valves to the environment.

3.1.5.1 Source Term

The source term for the CRE is assumed to result in fuel damage consisting of localized damage to fuel cladding with a limited amount of fuel melt occurring in the damaged rods. The source term for the CREA is described in RG 1.183, Appendix H, Regulatory Position 1, which

states that, "[a]ssumptions acceptable to the NRC staff regarding core inventory are in Regulatory Position 3 of this guide. For the rod ejection accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10 percent of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and the assumption that 100 percent of the noble gases and 25 percent of the iodines contained in that fraction are available for release from containment. For the secondary system release pathway, 100 percent of the noble gases and 50 percent of the iodines in that fraction are released to the reactor coolant."

The licensee assumed that as a result of the CRE, 10 percent of the fuel experiences departure from nucleate boiling (DNB) resulting in cladding damage. In addition, 2.5 percent of the damaged fuel also releases fuel melt activity.

Consistent with the guidance provided in RG 1.183, Appendix H, the licensee assumed that 10 percent of the core inventory of noble gases and iodine reside in the fuel gap. The licensee assumed that 12 percent of the core inventory of alkali metals, which include cesium and rubidium, reside in the fuel gap, as specified in Table 3 of RG 1.183. The licensee assumed that for the 10 percent of the fuel that experiences DNB, all of the gap activity contained in the affected fuel will be available for release in both the CRE-induced LOCA scenario and the secondary-side release scenario.

In accordance with RG 1.183, for the 0.25 percent of fuel experiencing fuel centerline melting (FCM), the licensee assumed that 100 percent of the noble gases, 25 percent of the iodines and 12 percent of the alkali metals in the affected fuel will be available for release from containment in the CRE-induced LOCA scenario. For the 0.25 percent of fuel experiencing FCM, the licensee assumed that 100 percent of the noble gases, 50 percent of the iodines and 12 percent of the of the alkali metals in the affected fuel will, be available for release from the RCS for the secondary-side CRE release scenario.

In accordance with RG 1.183, Appendix H, Regulatory Position 3, 100 percent of the released activity is assumed to be released instantaneously and mixed homogeneously throughout the containment atmosphere for the CRE-induced LOCA; and, 100 percent of the released activity is assumed to be released instantaneously and completely dissolved in the primary coolant and available for release to the secondary containment in the CRE secondary-side release scenario.

In accordance with RG 1.183, Appendix H, Regulatory Position 4, the licensee assumed that the chemical form of radioiodine released to the containment atmosphere consists of 95 percent CsI, 4.85 percent elemental iodine, and 0.15 percent organic iodide. After the application of the PC of 100, the effective release to the environment is then 82.7 percent particulate, 4.2 percent elemental and 13.1 percent organic. The licensee notes that this assumption results in a greater number of curies of iodines released than that required by RG 1.183 since it includes the release of particulates and no PC is used to reduce the amount of organics released.

The licensee assumed that the initial RCS iodine concentrations include a pre-existing iodine spike to the TS limit of 60 μCi/gm. This is a conservative treatment of the CRE analysis since RG 1.183 does not stipulate the inclusion of iodine spiking for accidents that result in fuel

damage. The licensee included the RCS cesium and rubidium concentrations corresponding to 1 percent FF. Initial RCS cesium and rubidium concentrations are not assumed to have spiked along with the iodines for the CREA dose consequence analysis. The licensee included the radiological dose contribution from the release of secondary coolant iodine activity at the TS limit of 0.1 μCi/gm DEI.

3.1.5.2 Transport from Containment

The licensee used the minimum containment free air volume to conservatively maximize the radioactive concentration in containment. In contrast, the licensee used the maximum containment free air volume for determining the containment leakage to conservatively maximize the value for containment leakage.

The licensee assumed that the activity released to the containment through the rupture in the reactor vessel head mixes instantaneously throughout the containment with no credit assumed for removal of iodine in the containment due to containment sprays or natural deposition. The licensee assumed that all containment leakage is at the TS limit of 0.3 percent per day for the first 24 hours and 0.15 percent per day thereafter.

The licensee assumed that iodines released to the containment from both the fuel and the RCS consist of 95 percent particulate, 4.85 percent elemental, and 0.15 percent organic per RG 1.183, Appendix H, Regulatory Position 4.

3.1.5.3 Transport from Secondary System

In accordance with RG 1.183, Appendix H, Regulatory Position 7, the licensee evaluated the transport of activity from the RCS to the SGs secondary side assuming a total primary-to-secondary leak rate equal to 1 gpm, which is higher than the TS total allowable leak rate of 0.42 gpm. The licensee assumed that this leak rate persists for a period of eight hours until shutdown cooling is in operation and the RCS and the SG pressures have equalized.

The licensee's leak rate testing results are adjusted so that the allowable leakage corresponds to a density of 8.33 lbm per gallon and accordingly this density was used to convert the volumetric leak rate to a total mass flow rate due to SG tube leakage in the CREA dose consequence analysis.

In accordance with RG 1.183, the licensee assumed that all noble gas radionuclides released from the primary system are released to the environment without reduction or mitigation. Following the guidance from RG 1.183, Appendix E, Regulatory Positions 5.5.1, 5.5.2, and 5.5.3, the licensee assumed that all of the primary-to-secondary leakage in both SGs mixes with the secondary water without flashing.

As discussed for the MSLB and the SGTR accidents, the licensee applied the guidance in Appendix E, Section 4 to the release of iodine from the SGs incorporating a PC of 100 for particulate and elemental iodine but not for organic iodine. When the PC of 100 is applied, as stated, to the chemical form of radioiodine released from the fuel, the resulting speciation for an SG release becomes 4.2 percent elemental iodine, 13.1 percent organic iodide, and

82.7 percent particulate iodine. As stated previously, this speciation is a conservative departure from the RG 1.183 guidance on the speciation for an SG release of 97 percent elemental iodine, 3 percent organic iodide with no particulate iodine released.

3.1.5.4 CR Ventilation Assumptions for the CRE

The licensee evaluated CR habitability for the CRE assuming that the CR ventilation system automatically transfers to the emergency mode of operation after the initiation of safety injection. The revised AST CR analyses assume that in the emergency mode all makeup flow is unfiltered. The total of 4 inches of charcoal filtration per train, which consists of 2 inches of the makeup filters and 2 inches of the cleanup filters, is not credited in the AST analysis. Only the recirculation filtration is credited with 95 percent removal of both elemental and organic iodine and 99 percent removal of particulates. Therefore, the assumed filtered make-up air flow is 0 cfm and the 2200 cfm make-up flow is added to the 100 cfm unfiltered in-leakage value. The 100 cfm unfiltered in-leakage values includes the contribution from door pumping action from CR ingress and egress. The AST analysis assumes that a total of 2300 cfm directly enters the CR without filtration. In addition, the licensee does not credit the use of non-ESF ventilation systems during the DBA analysis. The CR parameters used in the AST analyses are shown in Table 4 of this SE.

The licensee evaluated the radiological consequences resulting from the postulated CRE and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in SRP Section 15.0.1. The staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the staff are presented in Table 9 and the licensee's calculated dose results are given in Table 1. The staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The EAB, LPZ, and CR doses estimated by the licensee for the CRE were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.1.6 Locked Rotor Accident (LRA)

The accident considered begins with the instantaneous seizure of a reactor coolant pump (RCP) rotor which causes a rapid reduction in the flow through the affected RCS loop. The sudden decrease in core coolant flow causes a reactor trip on a low flow signal. The low coolant flow causes a degradation of core heat transfer, resulting in localized temperature and pressure changes in the core. The licensee's evaluation indicates that the fuel will experience a DNB which results in fuel cladding damage. Activity from the fuel cladding damage is transported to the secondary side due to primary-to-secondary side leakage. For conservatism, the licensee assumed a total primary-to-secondary leak rate equal to 1 gpm which is higher than the TS total allowable leak rate of 0.42 gpm. It is assumed that the LRA does not cause an increase in the magnitude of the pre-existing primary-to-secondary leakage.

3.1.6.1 Source Term

The licensee assumed that the instantaneous seizure of the RCP rotor associated with the LRA results in a small percentage of fuel clad damage. The dose analysis for this event conservatively assumes 10 percent fuel clad damage with no fuel melt predicted. Therefore, the source term available for release is associated with this fraction of damaged fuel cladding and the fraction of core activity existing in the gap. The licensee conservatively included iodine in the RCS due to a design-basis pre-accident 60 µCi/gm DEI spike. The licensee included the RCS noble gas activity associated with assumed 1 percent fuel defects. The licensee included the RCS cesium and rubidium concentrations corresponding to 1 percent FF, however, the cesium and rubidium concentrations are not assumed to spike along with the iodines.

The licensee incorporated release fractions and transport fractions that are consistent with RG 1.183, Appendix G, and Table 3. In accordance with RG 1.183, 5 percent of the core inventory of iodine and noble gas is assumed to be in the fuel-clad gap, with the exception of 8 percent assumed for iodine 131 and 10 percent assumed for krypton 85. Additionally, the licensee included 12 percent of the core cesium and rubidium as shown in Table 3 of RG 1.183.

The source term model also includes the maximum TS equilibrium secondary coolant activity concentration of 0.1 µCi/gm DEI.

3.1.6.2 Release Transport

The activity that originates in the RCS is released to the secondary coolant by means of the primary-to-secondary coolant leak rate. The licensee assumed a conservative value for the design-basis leak rate of 1.0 gpm. This 1.0 gpm leak rate is modeled as 0.65 gpm for the three SGs with tube coverage and 0.35 gpm for the SG with uncovered tubes. For the SG on the loop with the locked RCP rotor, the licensee assumed that the SG tubes become uncovered due to a feedwater isolation valve malfunction.

A LOOP is assumed to occur concurrently with the reactor trip which results in releases to the environment associated with the secondary coolant steaming from the SGs.

Because of the release dynamic of the activity from the SG PORVs, RG 1.183 allows for a reduction in the amount of activity released to the environment based on partitioning of nuclides between the liquid and gas states of water for this release path. For iodine, the PC of 100 was taken directly from the suggested guidance. Because of their volatility, 100 percent of the noble gases are assumed to be released.

The licensee assumed that the steaming release from the PORVs and primary-to-secondary coolant leakage end after 8 hours, at which time the RCS and the secondary system have reached pressure equilibrium. The licensee assumed that leakage via the MSIV above-seat drain orifices continues for 36 hours.

The licensee used the RADTRAD computer code to model the time dependent transport of radionuclides, from the primary-to-secondary side and consequently to the environment via the PORVs or MSSVs [main steam safety valve]. The licensee's analysis conforms with

Appendix G of RG 1.183 which identifies acceptable radiological analysis assumptions for an LRA. The licensee assumed the same CR ventilation timing sequence as was used for the FHA.

The licensee evaluated the radiological consequences resulting from the postulated LRA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident dose criteria specified in SRP 15.0.1. The staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the staff are presented in Table 10 and the licensee's calculated dose results are given in Table 1. The staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The EAB, LPZ, and CR doses estimated by the licensee for the LRA were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.2 Atmospheric Dispersion Estimates

The licensee used onsite meteorological data collected during calendar years 2000 through 2004 to generate new CR/TSC, EAB, and LPZ χ/Q values for the STP Unit 1 and Unit 2 dose assessments discussed above. The resulting χ/Q values represent a change from those used in the current STP UFSAR, Chapter 15, "Accident Analysis."

3.2.1 Meteorological Data

The 2000 through 2004 meteorological data were provided for staff review in the form of hourly meteorological data files in a format suitable for input into the ARCON96 atmospheric dispersion computer code, NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes," and in the form of a joint wind direction, wind speed and atmospheric stability distribution (joint frequency distribution, JFD) suitable for input into the PAVAN atmospheric dispersion computer code, NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations." The staff performed a quality review of the hourly meteorological databases using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data." Further review was performed using computer spreadsheets.

In the LAR dated March 22, 2007, the licensee stated that the data were gathered per RG 1.23, Rev. 0, "Onsite Meteorological Programs," and the meteorological tower was equipped with instrumentation that conforms to the system accuracy recommendations in RG 1.23 for the parameters used in this analysis. Wind speed and wind direction data were measured on the STP onsite 60-meter primary meteorological tower at a height of 10 meters above-ground level. Data were analyzed by a meteorologist to ensure reasonableness and consistency, as well as to verify the validity of the data. Wind data from the primary tower judged to be invalid were replaced with data from the 10-meter backup tower. When the Δ-T (change in temperature) instrument was inoperable, wind direction variance ($\sigma\theta$) data were used to maintain a 90 percent data recovery. RG 1.23 recommends a minimum data recovery rate of 90 percent. However, the ARCON96 and PAVAN computer codes used by the licensee in the atmospheric

dispersion analyses were written to use atmospheric dispersion categories derived from Δ-T measurements. As a result, NRC staff asked the licensee to quantify the amount of time from 2000 through 2004 that Δ-T data were substituted with σθ data and what adjustments, if any, were made to the σθ data to better simulate atmospheric stability categorization using the Δ-T methodology.

In its response by letter dated October 11, 2007, the licensee stated that σθ data were used approximately 6 percent of the time during the 2000 through 2004 interval and explained its procedure to adjust the σθ data. The licensee used the atmospheric stability categories defined in Table 2.5 of NUREG/CR-3332, "Radiological Assessment - A Textbook on Environmental Dose Analysis." If a stability class of unstable at night or stable during the day was estimated using σθ data, a neutral, "D," stability was assumed for that hour of data. This adjustment criterion is different than that footnoted with Table 2.5 which states that use of σθ to represent atmospheric stability when wind speeds are less than 1.5 meters per second should be substantiated. In a subsequent telephone conversation between the licensee and NRC staff, the licensee provided additional details of the procedure used to adjust the σθ data and stated that σθ data were used in place of invalid Δ-T data only during years 2003 and 2004. This could be inferred by examination of annotated raw data files that had been provided with the March 22, 2007, LAR submittal. NRC staff performed a cursory review of the electronic hourly data files to confirm that the 3-year period of data from 2000 through 2002 utilized Δ-T measurements only.

With respect to the reported 2000 through 2004 atmospheric stability measurements for STP, the duration of stable and unstable conditions were generally consistent with expected meteorological conditions. Further, stable and neutral conditions were generally reported to occur at night and unstable and neutral conditions during the day, with neutral or near-neutral conditions predominating during each year as expected. Unstable and stable conditions were reported to occur somewhat more frequently than typically observed at other nuclear power plant sites. The licensee noted the major local effect on the site meteorology resulting from its location near the Gulf of Mexico, including sea and land breeze circulations generated by temperature differences between the land and water.

The effect of the close proximity of the STP site to the Gulf of Mexico was also evident in the measured site winds from 2000 through 2004. During this time, winds were predominantly from the south, south southeast and southeast directions, with secondary flow clockwise from the north northwest through northeast directions. Winds from the west southwest and adjacent directions were very infrequent. As shown in Figure 2.3-2 in the STP UFSAR, the strong onshore and offshore flow between the Gulf of Mexico and Texas coastal area is regional in nature, with climatic wind distributions at Victoria, Corpus Christi and Galveston, Texas similar to those at the STP site. Wind speed and wind direction frequency distributions at STP for each measurement channel from 2000 through 2004 were reasonably similar from year to year. The NRC staff could not perform comparisons with winds at the 60-meter level since only 10-meter level wind data were provided as part of this LAR.

A comparison of the 2000 through 2004 10-meter level JFD derived by the NRC staff from the ARCON96 hourly data with the JFD developed by the licensee for input into the PAVAN atmospheric dispersion model showed good agreement. The licensee selected seven

wind-speed categories based upon Table 1 of RG 1.23. The selected wind-speed categories are 0.5, 3.5, 7.5, 12.5, 18.5, 24.5, 18.5, 24.5, and 36.0 miles per hour (mph). NRC staff judged that this choice of wind-speed categories resulted in some clustering of the data. In the March 22, 2007, LAR, the licensee acknowledged that RIS 2006-04 recommends that JFD input to PAVAN should have a large number of wind-speed categories at the lower wind speeds in order to produce the best results. The recommended wind-speed categories converted to mph are calm, 1.1, 1.7, 2.2, 2.8, 3.4, 4.5, 6.7, 8.9, 11.2, 13.4, 17.9, and 22.4 mph. However, the licensee judged the seven wind-speed categorization based upon RG 1.23 to be adequate for determining the offsite χ/Q values using PAVAN for this LAR. This is addressed further in Section 3.2.3.

The NRC staff has concluded that the 2000 through 2004 meteorological data measured at the STP site provide an acceptable basis for making atmospheric dispersion estimates for use in the dose assessments performed in support of this LAR. The conclusion is based upon the χ/Q values generated by the licensee with those generated by the NRC staff using current NRC guidance. The NRC staff used just the 2000 through 2002 meteorological data because these data are based upon Δ-T measurements only and do not include σθ data. In addition, when generating the comparison JFD, the NRC staff also binned the data to RIS 2006-04 wind-speed categories. The NRC staff notes that any future calculations of χ/Q values using ARCON96 or PAVAN should consider use of Δ-T based measurements only and that a large number of wind-speed categories at the lower wind speeds be used when data are binned into a JFD.

3.2.2 Control Room Atmospheric Dispersion Factors

The licensee used guidance provided in RG 1.194 to generate new CR χ/Q values. The licensee calculated χ/Q values for the LOCA, CREA, FHA, MSLB, SGTR, and LRA using the ARCON96 computer code. RG 1.194 states that ARCON96 is an acceptable methodology for assessing CR χ/Q values for use in DBA radiological analyses. The NRC staff evaluated the applicability of the ARCON96 model and concluded that there are no unusual siting, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude use of the ARCON96 model for the STP site.

The licensee executed ARCON96 using the 2000 through 2004 hourly data from the STP onsite meteorological tower. As noted in Section 3.2.1 above, stability class was generally calculated using Δ-T measurements between the 60-meter and 10-meter levels on the meteorological tower, occasionally supplemented in 2003 and 2004 with atmospheric stability estimates generated from σθ data. Based upon the facility configuration, the licensee provided six sets of χ/Q values, for releases from the Unit 1 and Unit 2 containment buildings, plant vents and PORVs to each unit's own CR air intake. Because the heights of all six of the postulated release locations are less than two and one-half times the height of adjacent buildings, they were modeled using the ARCON96 ground level release option in accordance with RG 1.194. The containment releases were modeled as diffuse sources and the other postulated releases were modeled as point sources. Postulated releases from the containment buildings were applied to the LOCA and CREA dose assessments. Releases from the plant vents were applied to the LOCA ESF leakage, LOCA supplemental purge and to the FHA in the FHB and containment building. Postulated releases from the east PORVs were applied to the MSLB, SGTR, and LRA dose assessments. In the October 11, 2007, supplemental submittal, the

licensee provided additional detailed information to justify that the most limiting release/receptor pair had been considered for each postulated DBA. CR χ/Q values were used as inputs for unfiltered in-leakage since the CRs are contained within the electrical auxiliary buildings and the intakes for the CRs and electrical auxiliary buildings are adjacent to each other for each respective unit.

For the purpose of calculating χ/Q values using the ARCON96 computer code, STP Unit 1 and Unit 2 are nearly identical "slide along" plants oriented on true north. Therefore, the distance, direction and height inputs to ARCON96 are essentially identical whether the postulated release is from Unit 1 to the Unit 1 CR intake or Unit 2 to the Unit 2 CR intake. The distances from each unit containment building, plant vent, and east PORV to its respective CR intake are approximately 62 meters, 63 meters, and 84 meters, respectively. The assumed containment, plant vent and east PORV release heights are about 31 meters, 21 meters, and 21 meters, respectively.

RIS 2006-04 states that when using ARCON96, two levels of wind speed and wind direction data should always be provided as input to each data file and fields of "nines" (e.g., 999) should be used to indicate invalid or missing data. In the calculations for this LAR, the licensee did not use upper level wind data, instead it left the data fields blank. The licensee attempted to code any hour with invalid 10-meter wind data as invalid by formatting the atmospheric stability as a 9. For this case, the NRC staff does not accept the licensee's deviation from RIS 2006-04 methodology, but has adopted the following approach in evaluating the acceptability of the licensee's results.

The NRC staff qualitatively reviewed the inputs to the ARCON96 calculations and found them generally consistent with site configuration drawings and staff practice except as noted above. Use of a single wind measurement height is acceptable for the calculations made for this LAR given that the heights of release are adequately near the 10-meter level. The NRC staff also generated random comparative χ/Q values using only the 2000 through 2002 hourly data which did not include σθ data and coded the upper level wind fields with nines to indicate that those data were missing. Based on its review approach, the NRC staff concludes that the resulting CR χ/Q values generated by the licensee and presented in Table 2 of this SE are acceptable for use in the accident dose assessments performed in support of this LAR. This conclusion is arrived at by a comparison of the χ/Q values generated by the licensee with those generated by the NRC staff, using the current NRC guidance. The NRC staff used just the 2000 through 2002 meteorological data, because those data are based upon Δ-T measurements only and do not include σθ data. In addition, when generating JFD, the NRC staff also binned the data into the RIS 2006-04 wind-speed categories.

3.2.3 Offsite Atmospheric Dispersion Factors

The licensee generated χ/Q values for the EAB and LPZ using the methodology described in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the PAVAN computer code. All releases were assumed to be ground level. The licensee input a building minimum cross-sectional area of 2734 square meters, a reactor building height of 61.9 meters and EAB and LPZ distances of 1430 meters and 4800 meters, respectively. The licensee used a JFD derived from the 2000 through 2004

10-meter wind data. As noted in Section 3.2.1 above, stability class was generally calculated using Δ-T measurements between the 60-meter and 10-meter levels on the meteorological tower, occasionally supplemented in 2003 and 2004 with stability estimates based upon σθ data. The JFD data were binned into seven wind-speed classes, 0.5, 3.5, 7.5, 12.5, 18.5, 24.5, 18.5, 24.5, and 36.0 mph based upon Table 1 of RG 1.23. However, NRC staff judged that this resulted in some clustering of the data. As a result, NRC staff asked the licensee to provide further justification that the use of these categories was adequate for determining the STP EAB and LPZ χ/Q values using the PAVAN computer code. In their response, by letter dated October 11, 2007, the licensee provided an assessment of the data using the alternative wind-speed binning suggested in RIS 2006-04 and provided a comparison of the resultant χ/Q values with those generated using the seven wind-speed classes. The licensee noted that the comparison showed that the results of the two binning schemes are essentially the same, with the χ/Q values resulting from the RG 1.23 wind-speed scheme as used in this LAR being slightly more conservative.

NRC staff qualitatively reviewed the inputs and assumptions to the licensee's PAVAN computer calculations and generated comparative χ/Q values. The staff used a JFD based upon only the 2000 through 2002 data which did not include σθ data and binned the data into the RIS 2006-04 wind-speed categories. Staff found the resultant χ/Q values to be similar to those calculated by the licensee and agrees with the licensee that the RG 1.23 wind-speed scheme as used in this LAR is slightly more conservative. However, NRC staff also noted that other calculated values not used in this LAR were slightly less conservative. Therefore, on the basis of this review, the staff has concluded that the resulting EAB and LPZ χ/Q values generated by the licensee and presented in Table 2 of this SE are acceptable for use in the accident dose assessments performed in support of this LAR. However, NRC staff notes that any future calculations of χ/Q values using PAVAN should consider use of Δ-T based measurements only and a large number of wind-speed categories at the lower wind speeds.

3.2.4 Containment Sump pH

The licensee's application for AST amendments included the analyses the containment sump pH. This analysis was performed to ensure that the particulate iodine generated in the damaged core and deposited into the containment sump water during the DBA LOCA does not re-evolve beyond the amount recognized in the DBA LOCA analysis. The objective of this analysis is to determine the transient containment sump pH so that the removal of elemental and particulate iodine from the containment atmosphere in course of DBA LOCA could be properly evaluated. The licensee based its calculation of pH on the methodology developed in NUREG/CR-5950 (Iodine Evolution and pH Control). This methodology was incorporated in the STARpH computer code (A Code for Evaluation Containment Water Pool pH during Accidents) which was used by the licensee for performing its numerical calculation. Although this code was not accessible to the NRC staff, the licensee provided enough information on the code to permit the staff to perform an independent evaluation. Also, this code has been previously used to perform pH evaluations approved by the NRC staff in other plants.

The licensee postulated that after a LOCA the radioactive iodine is released from the damaged core to the containment. Most of this iodine is in ionic form (I-) easily soluble in the sump water. However, some of it will be converted into molecular form (I2) which is scarcely soluble in water.

Some of this iodine could be released to the outside environment contributing to the radiation dose rates. Since the conversion of iodine from ionic to molecular form is pH dependent, RG 1.183 stipulates that the sump water pH after a LOCA should be maintained at the value equal or higher than 7 to prevent conversion of the ionic iodine to the less soluble molecular form.

The post-LOCA sump water pH is determined by the dissolved chemicals and the buffer. In the submittal, the licensee listed the dissolved chemicals which will affect the sump water pH:

- boric acid from the reactor coolant system
- boric acid from the accumulators
- boric acid from the refueling water storage tank (RWST)
- hydrochloric acid
- nitric acid

This list is not all-inclusive. There are some other chemicals, including cesium hydroxide, which are not included in the list because in the licensee's opinion their effect on the sump pH was insignificant. Boric acid was added to the sump at the beginning of a LOCA and the other acids were generated in the containment throughout the entire 30-day period. These chemicals are responsible for a continuous decrease of the sump pH. The hydrochloric acid was generated by decomposition in a radiation field of the Hypalon cable insulation containing chlorine. Nitric acid was produced by irradiation of water and air in the containment. In the letters dated March 22, 2007 (ADAMS Accession No. ML070890473), and July 18, 2007 (ADAMS Accession No. ML072050341), the licensee described the procedures for determining the concentrations of these acids in the sump water.

In order to counteract the effect of strong acid the licensee buffers the sump water by adding 11,500 pounds of trisodium phosphate dodecahydrate (TSP). The TSP is kept in steel baskets and is dissolved when enough water accumulates in the sump. However, the licensee's analysis has determined that the amount of TSP is insufficient to maintain the sump water pH equal or higher than 7 for the entire 30-day period. Due to the excess of strong acids (hydrochloric and nitric), the sump pH will be maintained at a value at least 7 for less than 3 days after a LOCA, then it will drop to 6.9 and eventually to 6.8 at 30 days after a LOCA.

Although the results of this analysis indicate that the sump pH will not remain above a value of 7 for 30 days, as stipulated by RG 1.183, the licensee concluded that there should be little impact on the actual iodine re-evolution due to the following conservatism in the analysis:

- overestimated mass of cable insulation produces excess of hydrochloric acid
- cesium hydroxide is not credited in long-term pH analyses and the determination of final pH value

- no credit is taken for basic alkali metal compounds that result from fission products co-released with the iodine

- conservative assumptions were made to retain 10 percent of non-noble gas activity as airborne activity for the full 30 days

- conservative assumptions were made concerning the vulnerability of cables to beta radiation

The staff concurs with the licensee that in light of the conservatisms of the above analysis, the small departures from the criteria of maintaining the post-LOCA sump pH to level equal to or greater than 7 is acceptable.

The NRC staff reviewed the licensee's analyses, and justifications provided by the licensee. Based on that review, the NRC staff concludes that the licensee's analysis satisfactorily demonstrates that the pH of water in the containment sump during the 30-day post-LOCA period will not cause significant amount of iodine to convert into the volatile molecular form.

3.3 Conclusions of the NRC Staff Evaluation of Licensee's Radiological Dose Assessments

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of DBAs with full implementation of AST at STP, Units 1 and 2. The NRC staff concludes that the licensee used methods of analysis and assumptions consistent with the conservative regulatory requirements and guidance described in Section 2.0 above. The NRC staff compared the doses estimated by the licensee to the applicable dose guidelines and criteria referenced in Section 2.0, "Regulatory Evaluation." Based on that comparison, the NRC staff concludes that there is a reasonable assurance that the licensee's estimates of the EAB, LPZ, and CR doses comply with the regulatory requirements. The NRC staff also concludes that there is reasonable assurance that STP, Units 1 and 2 as modified by the requested license amendments, will continue to provide sufficient safety margins and adequate defense-in-depth, under conditions of unanticipated events, and in presence of the uncertainties in accident progression, assumptions, parameters, and analyses outlined above. Therefore, the proposed changes to the licensing basis are acceptable with respect to the radiological consequences of DBAs.

With this approval, the previous accident source term in the STP, Units 1 and 2 licensing basis is superseded by the revised licensing basis, incorporating the AST as proposed by the licensee. The previous offsite and CR accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE guidelines and criteria of 10 CFR 50.67, or fractions thereof, as defined in RG 1.183. All future radiological accident analyses performed to show compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as defined in the STP, Units 1 and 2 design basis, and modified by the present amendments.

3.4 Technical Specification Changes

3.4.1 Section 1.0, Definitions

The licensee has proposed to revise the dose conversion factors (DCFs) used to calculate the dose from DEI concentration to those listed in Federal Guidance Report 11 (FGR No. 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," 1988; Table 2.1, Exposure-to-Dose Conversion Factors for Inhalation, instead of Table E-7 of NRC RG 1.109, Revision 1, October 1977. The licensee has proposed to drop "thyroid" from the definition of dose to reflect that the dose is now the TEDE, based on AST methodology.

The intent of the TS on RCS specific activity is to ensure that assumptions made in the DBA radiological consequence analyses remain bounding. As such, the specification should have a basis consistent with the basis of the dose analyses. The licensee currently calculates DEI using thyroid DCFs, since the limiting analysis result was the thyroid dose. The AST analyses, however, determine the TEDE, rather than the whole body dose and thyroid dose as done previously. Therefore, the licensee proposes to use the inhalation CEDE DCFs from FGR No. 11, to calculate DEI. The NRC staff has evaluated the proposed definition of DEI and has determined that the incorporation of the CEDE DCFs from FGR No. 11 in the DEI definition is acceptable.

3.4.2 Functional Unit 3.b.4 of Table 3.3-3, ESF Actuation System Instrumentation

The licensee proposes to delete Modes 5 and 6 for Functional Unit 3.b.4, "Containment Ventilation Isolation RCB Purge Radioactivity - High," as Applicable Modes, and has modified ACTION 18 appropriately, because it has determined that automatic isolation is no longer required during core alterations or movement of irradiated fuel within containment to meet the STP, Units 1 and 2 design basis incorporating the AST in accident analysis.

The NRC staff reviewed the proposed change. In Section 4.4, “Fuel Handling Accident,” of the LAR, the licensee concludes that of the accidents postulated to occur during core alterations (i.e., inadvertent criticality, FHA and the loading of a fuel assembly or control component in the wrong location) only the FHA results in cladding damage and potential radiological release.

The analysis of Section 4.4 assumed at least 42 hours had elapsed since the fuel was last critical. The proposed amendment takes credit for the normal decay of irradiated fuel rather than crediting certain active mitigation systems. Containment isolation was not a requirement in the analysis. Neither FHB nor RCB filtration was credited in the analysis. The FHA Dose Results of Table 4.4.3 of the AST analysis for the EAB, LPZ, CR, and TSC indicate that the doses at each receptor (rem TEDE) are substantially below reference values set forth in 10 CFR 50.67 and the guidance of RG 1.183.

The operability of “Containment Ventilation Isolation RCB Purge Radioactivity – High” instrumentation for actuation on RCB Purge isolation is no longer required during Modes 5 and 6 for the reasons indicated above. For Modes 5 and 6 of STP, Units 1 and 2, this instrumentation is no longer part of a primary success path that actuates to mitigate a DBA that

either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Accordingly, for these modes of plant operation, the instrumentation no longer satisfies the criteria for TS inclusion per 10 CFR 50.36(d)(2)(ii).

Consistent with the guidance contained in Revision 2 of TSTF-51, Attachment 5 of the LAR contains the necessary commitments (Commitment #s 3 and 6) to isolate the RCB for postulated FHA in the RCB, to further reduce dose by natural decay and to enable the ventilation system to draw the release from a postulated FHA in the containment in the proper direction such that it can be monitored (not treated). These commitments will further reduce dose below those “Dose (0-2 hour)” values listed in Table 4.4.3 of the AST analysis.

Based on the above, the NRC staff finds the proposed changes to Functional Unit 3.b.4 of Table 3.3-3 acceptable.

3.4.3 Functional Unit 10 of Table 3.3-3, ESF Actuation System Instrumentation

The licensee has proposed to: (1) delete Modes 5 and 6 for Functional Unit 10.a, c, and d, "Control Room Ventilation," as Applicable Modes; (2) modify the Action 28 requirement for Functional Unit 10.d, "Control Room Intake Air Radioactivity – High," to delete suspension of core alterations, movement of irradiated fuel, and crane operation with loads over the spent fuel pool; (3) modify Action 28 in Table 3.3-3, item b. to read, “[w]ith the number of OPERABLE channels two less than the Minimum Channels OPERABLE requirement, within 12 hours initiate and maintain operation of the CR Makeup and Cleanup Filtration System (at 100 percent capacity) in the recirculation and makeup filtration mode.”

These changes are proposed because the accident mitigation capabilities of this system are no longer credited in AST DBA analysis for activities performed during these Modes.

The staff reviewed the proposed change. In Section 4.4, “Fuel Handling Accident,” of the LAR, the licensee concludes that of the accidents postulated to occur during core alterations (i.e., inadvertent criticality, FHA and the loading of a fuel assembly or control component in the wrong location) only the FHA results in cladding damage and potential radiological release.

The analysis of Section 4.4 was performed assuming at least 42 hours had elapsed since the fuel was last critical. The proposed amendment takes credit for the normal decay of irradiated fuel rather than crediting certain active mitigation systems. CR system filtration was not credited in the analysis. The FHA Dose Results of Table 4.4.3 from the AST analysis for the CR indicate that the CR dose (rem TEDE) is substantially below reference values set forth in 10 CFR 50.67 and the guidance of RG 1.183.

The operability of “Control Room Ventilation” instrumentation for “Manual Initiation,” “Automatic Actuation Logic and Actuation Relays,” and “Control Room Intake Air Radioactivity – High,” is no longer required during Modes 5 and 6 since the accident mitigation capabilities of the CR system filtration was not credited in the AST FHA DBA. For Modes 5 and 6, this instrumentation is no longer part of a primary success path that actuates to mitigate a DBA that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Accordingly, for

these modes of plant operation, the instrumentation no longer satisfies the criteria for TS inclusion as required by 10 CFR 50.36(d)(2)(ii).

In summary, the NRC staff finds the proposed changes to Functional Unit 10 of Table 3.3-3 and Action 28 acceptable.

Whenever irradiated fuel is being moved or when conducting crane operation with loads over the spent fuel pool, the licensee commits in Attachment 5 of the LAR:

1. to keep the Control Room Makeup and Cleanup Filtration System OPERABLE or to restore at least one train to an OPERABLE status within two hours;

2. to keep radiation monitoring instrumentation functional to ensure that a release following a Fuel Handling Accident is monitored; and

3. to place at least one train of the Control Room Makeup and Cleanup Filtration System into operation within two hours of a Fuel Handling Accident.

This is consistent with the guidance contained in TSTF-51, Revision 2. In the event of an FHA these commitments will further reduce dose to the occupants of the CR below the Control Room "Dose (0-2 hour)" value listed in Table 4.4.3 of the AST analysis.

Modification of Action 28 will delete reference to suspension of core alterations, movement of irradiated fuel, and crane operation with loads over the spent fuel pool in the action statements (i.e., of Action 28). The deletion of these action statements is acceptable because the accident mitigation capabilities of the Control Room Ventilation system are no longer credited in AST DBA analysis during these activities.

In Supplement 2 to the LAR, the modified Action 28-b. recaptures the existing 12 hours allowed to initiate and maintain operation of the Control Room Makeup and Cleanup Filtration System (at 100 percent capacity) in the recirculation and makeup filtration mode. This allowed time already exists in the current TS. This change was not captured in the original LAR submittal and is editorial in nature. The staff finds this change acceptable.

3.4.4 Functional Unit 11 of Table 3.3-3, ESF Actuation System Instrumentation

The licensee requested changes to delete the requirement for an operable Functional Unit 11 (a, b, c, and d), FHB HVAC, actuation instrumentation; because "the accident mitigation capabilities of the FHB HVAC system are no longer credited in AST DBA analysis."

The NRC staff reviewed the proposed change. The proposed change deletes from TS Table 3.3-3 the operability requirements for the FHB HVAC system's actuation instrumentation. The staff reviewed the licensee's AST DBA analyses for LOCA, FHA, MSLB, SGTR, CREA, and LRA. The accident mitigation capabilities of the FHB HVAC filter system are no longer credited in any of the design bases of those accident analyses.

For all modes of STP, Units 1 and 2 operation, this instrumentation is no longer part of a primary success path that actuates to mitigate a DBA that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Accordingly, for these modes of plant operation, the instrumentation no longer satisfies the criteria for TS inclusion required by 10 CFR 50.36(d)(2)(ii).

Whenever irradiated fuel is being moved or when conducting crane operation with loads over the spent fuel pool the licensee commits in Attachment 5 of the LAR:

1. to keep as OPERABLE, or to restore at least one train of, the Fuel Handling Building Exhaust Air to an OPERABLE status within two hours;

2. to keep radiation monitoring instrumentation functional to ensure that a release following an FHA is monitored; and

3. to place at least one train of the FHB Exhaust Air system into operation within two hours of an FHA.

This is consistent with the guidance contained in TSTF-51, Revision 2. The purpose of these commitments is to maintain the FHB Ventilation System and the associated radiation monitoring available to reduce doses even further below that provided by natural decay and to avoid unmonitored releases; and to enable the FHB Ventilation System to draw the release from a postulated FHA in the FHB in the proper direction such that it can be treated and monitored.

In summary, the NRC staff finds the proposed changes to Functional Unit 11 of Table 3.3-3 acceptable.

3.4.5 Action 20 to Table 3.3-3, ESF Actuation System Instrumentation

The licensee had proposed an administrative change to remove a Note from ACTION 20 because the provisions of the Note had expired.

The Note for Action 20 had a provision that read “[t]his provision will expire 30 days after approval of the amendment.” This Note was incorporated into Amendment No. 176, which was approved by the NRC on January 11, 2007 (ADAMS Accession No. ML070100198), and was specific to STP, Unit 1. However, since the Note for Action 20 was removed and incorporated in Amendment Nos. 179 and 166, issued on July 13, 2007 (ADAMS Accession No. ML071780186), the licensee’s proposed request is no longer necessary.

3.4.6 Table 3.3-4, Engineered Safety Features Actuation System Instrumentation Trip Setpoints

The licensee has proposed to delete the trip setpoints and allowable values for Functional Unit 11, "FHB HVAC,” because the accident mitigation capabilities of this system are no longer credited in AST DBA analyses.

The NRC staff reviewed the proposed change. Since the operability requirements for the FHB HVAC system's actuation instrumentation are removed from the TS, the NRC staff finds that the deletion of the associated TS trip setpoints of Table 3.3-4 is acceptable.

3.4.7 Table 4.3-2 (Functional Unit 3.b.4), Engineered Safety Features Actuation System Instrumentation Surveillance Requirements

The licensee proposed to delete Modes 5 and 6 for Functional Unit 3.b.4, "Containment Ventilation Isolation RCB Purge Radioactivity - High," as APPLICABLE MODES, because automatic isolation is no longer required during core alterations or movement of irradiated fuel within containment to meet the AST DBA analysis.

The NRC staff reviewed the proposed change. In Section 4.4, "Fuel Handling Accident," of the LAR, the licensee concludes that of the accidents postulated to occur during core alterations (i.e., inadvertent criticality, FHA and the loading of a fuel assembly or control component in the wrong location) only the FHA results in cladding damage and potential radiological release.

The analysis of Section 4.4 was performed assuming at least 42 hours had elapsed since the fuel was last critical. The proposed amendment takes credit for the normal decay of irradiated fuel rather than crediting certain active mitigation systems. Containment isolation was not a requirement in the analysis. Neither FHB nor RCB filtration was credited in the analysis. The FHA Dose Results of Table 4.4.3 of the AST analysis for the EAB, LPZ, CR, and TSC indicate that the doses at each receptor (rem TEDE) are substantially below regulatory reference values in 10 CFR 50.67 and RG 1.183 guidance.

Based on the above, the NRC staff concludes that the operability of the "Containment Ventilation Isolation RCB Purge Radioactivity – High" instrumentation for actuation on RCB Purge isolation is no longer required during Modes 5 and 6. It follows that the need, to ensure that the LCOs are satisfied through surveillance requirements (SRs) in Modes 5 and 6 no longer exists. For Modes 5 and 6 the instrumentation identified in Functional Unit 3.b.4 in Table 4.3-2 is no longer part of a primary success path that actuates to mitigate a DBA that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Accordingly, for these modes of plant operation, the instrumentation no longer satisfies the criteria for TS inclusion required by 10 CFR 50.36(d)(2)(ii).

Consistent with the guidance contained in Revision 2 of TSTF-51, Attachment 5 of the LAR contains the necessary commitments (Commitment #s 3 and 6) to isolate the RCB for postulated FHA in the RCB and further reduce dose by natural decay and to enable the ventilation system to draw the release from a postulated FHA in the containment in the proper direction such that it can be monitored (not treated). These commitments will further reduce dose below those "Dose (0-2 hour)" values listed in Table 4.4.3 of the AST analysis.

Based on the above, the NRC staff finds the proposed changes to Functional Unit 3.b.4 in Table 4.3-2 acceptable.

3.4.8 Table 4.3-2 Functional Unit 10, Engineered Safety Features Actuation System Instrumentation Surveillance Requirements

The licensee has proposed to delete Modes 5 and 6 for Functional Unit 10, "Control Room Ventilation," as Applicable Modes, because the accident mitigation capabilities of this system are no longer credited in AST DBA analysis for activities performed during these Modes.

The NRC staff reviewed the proposed change. In Section 4.4, “Fuel Handling Accident,” of the LAR, the licensee concludes that of the accidents postulated to occur during core alterations (i.e., inadvertent criticality, FHA and the loading of a fuel assembly or control component in the wrong location) only the FHA results in cladding damage and potential radiological release.

The analysis of Section 4.4 was performed assuming at least 42 hours had elapsed since the fuel was last critical. The proposed amendment takes credit for the normal decay of irradiated fuel rather than crediting certain active mitigation systems. CR system filtration was not credited in the analysis. The FHA Dose Results of Table 4.4.3 of the AST analysis for the CR indicate that the CR dose (rem TEDE) is substantially below reference values set forth in 10 CFR 50.67 and the guidance of RG 1.183.

Based on the above, the NRC staff concludes that the operability of the “Control Room Ventilation” actuation instrumentation is no longer required during Modes 5 and 6. It follows that the need to ensure that the LCOs are satisfied through SRs in Modes 5 and 6 no longer exists. For Modes 5 and 6, the instrumentation identified in Functional Unit 10 in Table 4.3-2 is no longer a part of a primary success path that actuates to mitigate a DBA that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Accordingly, for these modes of plant operation, the instrumentation no longer satisfies the criteria for TS inclusion pursuant to requirements of 10 CFR 50.36(d)(2)(ii).

Based on the above, the NRC staff finds acceptable the proposed changes to Functional Unit 10 in Table 4.3-2 of the AST analysis.

3.4.9 Table 4.3-2, Engineered Safety Features Actuation System Instrumentation Surveillance Requirements

The licensee has proposed to delete the requirement for performing surveillances for Functional Unit 11, Fuel Handling Building (FHB) Heating, Ventilation and Air Conditioning (HVAC) actuation instrumentation, because the accident mitigation capabilities of the FHB HVAC system are no longer credited in AST DBA analysis.

The NRC staff reviewed the proposed change. The staff reviewed the licensee’s AST DBA analyses for LOCA, FHA, MSLB, SGTR, CREA, and LRA. The accident mitigation capabilities of the FHB HVAC filter system are no longer credited in any of these DBA analyses.

The proposed changes addressed in Section 3.4 of this SE will delete from TS Table 3.3-3 the operability requirements for the FHB HVAC system’s actuation instrumentation. It follows that the need to ensure that LCOs are satisfied through SRs no longer exists. For Modes 1, 2, 3, and 4 of operation or with irradiated fuel in the spent fuel pool, the instrumentation identified in

Functional Unit 11 in Table 4.3-2 is no longer part of a primary success path that actuates to mitigate a DBA that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Accordingly, the instrumentation no longer satisfies the criteria required by 10 CFR 50.36(d)(2)(ii) for TS inclusion.

Based on the above, the NRC staff finds the proposed changes to Functional Unit 11 in Table 4.3-2 acceptable.

3.4.10 TS 3/4.7.7, Control Room Makeup and Cleanup Filtration System

The licensee has proposed to:

- Delete the Applicability of Modes 5 and 6 to Limiting Condition for Operation (LCO) 3.7.7, to suspend all operations during core alterations, movement of irradiated fuel, and crane operation with loads over the spent fuel pool; "because the accident mitigation capabilities of this system are no longer credited in AST design basis accident analysis during these activities." Delete the requirements to suspend operations involving positive reactivity additions that could result in loss of required Shutdown Margin or required boron concentration, because, "adequate Shutdown Margin is controlled by TS 3/4.1.1, 'Shutdown Margin Boration Control' and adequate boron concentration is controlled by TS 3/4.9.1, 'Boration Concentration.' The safety analysis concludes that administrative controls and operator response time are adequate measures to preclude a loss of required Shutdown Margin or required boron concentration. In addition, requirements to suspend operations involving positive reactivity additions are not found in Standard Technical Specifications for control room ventilation systems. Thus, there are no radiological consequences."

- Modify Action C for Modes 1, 2, 3, and 4 to delete the requirements to suspend all operations involving movement of spent fuel and crane operations with loads over the spent fuel pool because the accident mitigation capabilities of this system are no longer credited in AST design-basis accident analysis.

- Delete a footnote from surveillance requirement 4.7.7.e 3) because the provisions of the footnote have expired.

The NRC staff reviewed the proposed changes. In Section 4.4, "Fuel Handling Accident," of the LAR, the licensee concludes that of the accidents postulated to occur during core alterations (i.e., inadvertent criticality, FHA and the loading of a fuel assembly or control component in the wrong location) only the FHA results in cladding damage and potential radiological release.

The analysis of Section 4.4 was performed assuming at least 42 hours had elapsed since the fuel was last critical. The proposed amendment takes credit for the normal decay of irradiated fuel rather than crediting certain active mitigation systems. CR system filtration was not credited in the analysis. The FHA dose results of Table 4.4.3 from the AST analysis for the CR indicate that the CR dose (rem TEDE) is substantially below the reference values set forth in 10 CFR 50.67 and the guidance of RG 1.183.

The operability of Control Room Makeup and Cleanup Filtration System is no longer required during Modes 5 and 6 since the accident mitigation capabilities of the CR system filtration was not credited in the AST design-basis analyses. For Modes 5 and 6, the CR Makeup and Cleanup Filtration System is no longer part of a primary success path that actuates to mitigate a DBA that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Accordingly, for Modes 5 and 6 of plant operation, the Control Room Makeup and Cleanup Filtration System no longer satisfies the criteria for requiring Technical Specification pursuant to 10 CFR 50.36(d)(2)(ii).

In summary, the staff finds the proposed changes to TS 3/4.7.7 acceptable.

During the activities of core alterations, movement of irradiated fuel or crane operation with loads over the spent fuel pool, the licensee commits in Attachment 5 of the LAR:

- to keep the Control Room Makeup and Cleanup Filtration System OPERABLE or to restore at least one train to an OPERABLE status within two hours;
- to keep radiation monitoring instrumentation functional to ensure that a release following an FHA is monitored; and
- to place at least one train of the Control Room Makeup and Cleanup Filtration System into operation within two hours of an FHA.

This is consistent with the guidance contained in TSTF-51, Revision 2. In the event of an FHA, these commitments will further reduce dose to the occupants of the CR below the CR "Dose (0-2 hour)" value listed in Table 4.4.3 of the AST analysis.

To meet the above commitments the licensee includes in the LAR planned changes to STP, Units 1 and 2 Technical Requirements Manual (TRM) to keep the Control Room Makeup and Cleanup Filtration System in a near operable status ("i.e. restore to Operable within 2 hours").

The licensee stated that TS 3/4.1.1, "Boration Control," is applicable to MODES 1, 2, 3, 4, and 5 and provides LCOs to ensure that Shutdown Margin is within limits provided in the Core Operating Limits Report (COLR). Also, TS 3/4.9.1 is applicable to MODE 6 and provides LCOs to ensure boron concentration of the reactor coolant system and the refueling canal is maintained uniform and sufficient to ensure that restrictive reactivity conditions are met.

The NRC staff reviewed the licensee's request to Modify ACTION C for MODES 1, 2, 3, and 4 to delete the requirements to suspend all operations involving movement of spent fuel and crane operations with loads over the spent fuel pool, because the accident mitigation capabilities of this system are no longer credited in AST DBA analysis.

For the activities of movement of spent fuel and crane operation with loads over the spent fuel pool during Modes 1, 2, 3, and 4, the Operability of the Control Room Makeup and Cleanup Filtration System is no longer required. The accident mitigation capabilities of the CR system filtration was not credited in the FHA design-basis AST analysis. For these particular activities

during Modes 1, 2, 3, and 4, the Control Room Makeup and Cleanup Filtration System is no longer part of a primary success path that actuates to mitigate an FHA DBA that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Accordingly, pursuant to requirements of 10 CFR 50.36(d)(2)(ii), during the above-discussed activities of movement of spent fuel and crane operation with loads over the spent fuel pool in Modes 1, 2, 3, and 4, the SSCs that comprise the Control Room Makeup and Cleanup Filtration System do not satisfy the criteria for inclusion in TS.

Based on this, the NRC staff finds that the proposed changes to Action C for Modes 1, 2, 3, and 4 of TS 3/4.7.7 acceptable.

The NRC staff reviewed the licensee's request to remove the footnote from SR 4.7.7.e.3. This is an administrative change, because the provisions of the footnote have expired.

The footnote is located on page 3/4 7-18 of the current STP, Units 1 and 2, and reads:

1. Measured points at a positive pressure but less than 1/8 inch Water Gauge are acceptable if an evaluation, considering appropriate compensatory action, demonstrates that the condition meets the requirements of GDC 19. The provisions of this note expire at 0800 on September 19, 2005.

2. The [NRC] staff's safety evaluation report will be issued after September 19, 2005. Therefore, the applicability of this footnote has expired. Accordingly, the NRC staff finds that the removal of the footnote is acceptable.

The licensee proposed to delete SR 4.7.7.a to verify at least once per 12 hours that the CR air temperature is less than or equal to 78 ºF during Mode 5 and Mode 6.

The NRC staff reviewed the proposed change. This proposal will result in relocation of the SR from TS to the TRM. The modified TRM will include an SR to verify at least once per 12 hours that the CR air temperatures are less than or equal to 78 ºF during the movement of irradiated fuel or when conducting crane operation with loads over the spent fuel pool. If this SR is not met, the licensee must suspend all operations involving movement of irradiated fuel and crane operation with loads over the spent fuel pool.

The licensee stated that:

> The proposed TRM requirement ensures that the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation in the CR and also ensures a habitable environment for CR personnel in the event of an FHA.

The licensee also stated that:

> Control room temperature limits are routinely met with one Control Room Makeup and Cleanup Filtration System operating with cooling provided to the system air handling unit by the same train Essential Chilled Water System. In the event of an FHA, the heat load

in the control room is not affected such that one Control Room Makeup and Cleanup Filtration System is sufficient to meet temperature limits.

The staff compared the current TS SR for monitoring CR temperature against the four criteria for TS inclusion pursuant to 10 CFR 50.36(d)(2)(ii). Based on this comparison, the staff concludes that the CR temperature monitoring requirement meets none of the criteria for requiring TS inclusion pursuant to 10 CFR 50.36(d)(2)(ii) and can be relocated to the TRM.

Based on above evaluation, the NRC staff finds the proposed changes to SR 4.7.7.a acceptable.

3.4.11 TS 3/4.7.8, Fuel Handling Building (FHB) Exhaust Air System

The licensee proposed to delete TS 3/4.7.8 in its entirety. The licensee stated in the LAR that the accident mitigation capabilities of the FHB Exhaust Air HVAC system are no longer credited in AST DBA analysis.

The staff reviewed the proposed change. The staff reviewed the licensee's AST DBA analyses for LOCA, FHA, MSLB, SGTR, CREA, and LRA. The accident mitigation capabilities of the FHB HVAC filter system are no longer credited in any of these DBA analyses.

Whenever irradiated fuel is being moved or when conducting crane operation with loads over the spent fuel pool, the licensee commits in Attachment 5 of the LAR:

- to keep as OPERABLE or to restore at least one train of the Fuel Handling Building Exhaust Air to an OPERABLE status within two hours;
- to keep radiation monitoring instrumentation functional to ensure that a release following an FHA is monitored; and
- to place at least one train of the FHB Exhaust Air system into operation within two hours of an FHA.

This is consistent with the guidance contained in TSTF-51, Revision 2. The purpose of these commitments is to maintain the FHB Ventilation System and the associated radiation monitoring available to reduce doses even further below that provided by natural decay, to avoid unmonitored releases, and to enable the FHB Ventilation System to draw the release from a postulated FHA in the FHB in the proper direction such that it can be treated and monitored.

To meet the above commitments, the licensee includes in the license amendment submittal planned changes to STP, Units 1 and 2's TRM to keep the FHB Exhaust Air System in a near operable status ("i.e., restore to Operable within two hours") and to keep the radiation monitoring instrumentation functional during the movement of irradiated fuel or when conducting crane operation with loads over the spent fuel pool.

For all Modes of STP, Units 1 and 2 operation, the FHB Exhaust Air System is no longer part of a primary success path that actuates to mitigate a DBA that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Accordingly, pursuant to 10 CFR 50.36(d)(2)(ii), for the stated modes of plant operation, the SSCs of the FHB Exhaust Air System no longer satisfy the criteria for requiring inclusion in the TS.

In summary, the NRC staff finds acceptable the proposed change to delete TS 3/4.7.8 in its entirety.

3.4.12 TS 3.8.1.2, AC Sources Shutdown

The licensee proposed to delete the actions to suspend movement of irradiated fuel or crane operation with loads over the spent fuel pool if the LCO is not met.

The licensee stated in the LAR that the FHA analysis no longer credits the mitigation systems that are dependent upon AC Sources Shutdown electrical power.

The Applicability for TS 3.8.1.2 reads, “MODE 5 and MODE 6 with water level in the refueling cavity < 23 feet above the reactor pressure vessel flange.”

The NRC staff reviewed the licensee’s AST DBA analyses for FHA. Assumption 5 of Section 4.4.5 of the LAR reads “[a] water depth above the damaged fuel of 23 feet is the limiting case.” This conflicts with the above Applicability statement.

This conflict was reported to the licensee by the NRC staff. In its December 18, 2007, letter, the licensee agreed that an STP, Units 1 and 2 TS conflict exists, and responded by withdrawing this specific request.

3.4.13 TS 3.8.1.3, AC [Alternating Current] Sources Shutdown

The licensee proposed to delete the actions to suspend movement of irradiated fuel or crane operation with loads over the spent fuel pool if the Limiting Condition for Operation is not met.

The licensee stated in the LAR that it proposes to delete the actions, because the FHA analysis no longer credits the mitigation systems that are dependent upon this source of electrical power. The NRC staff reviewed the proposed change. The Applicability for TS 3.8.1.3 reads, “MODE 6 with water level in the refueling cavity ≥ 23 feet above the reactor pressure vessel flange.”

The NRC staff reviewed the licensee's AST DBA analyses for FHA. The NRC staff noted that the accident mitigation capabilities of the following SSCs:

1. FHB Exhaust Air System and its actuation instrumentation;

2. Control Room Makeup and Cleanup Filtration System and its actuation instrumentation; and

3. Containment Ventilation Isolation RCB Purge Radioactivity – High actuation instrumentation (i.e., on RCB Purge isolation)

are no longer credited in the design-basis FHA accident analysis.

For shutdown Mode 6, it follows that the relevant portions of the AC electrical power sources used in support of these SSCs' mitigation capabilities are no longer required during movement of irradiated fuel or during crane operation with loads over the spent fuel pool.

For Mode 6 of TS 3.8.1.3 and during movement of irradiated fuel and/or during crane operation with loads over the spent fuel pool, the relevant portions of the AC electrical power sources used in support of these SSCs are no longer a part of a primary success path used to mitigate a DBA that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. For the relevant portions of the AC electrical power sources used in support of these SSCs, the criteria for TS inclusion per 10 CFR 50.36(d)(2)(ii) are no longer met.

Based on the above, the NRC staff finds the proposed change acceptable.

3.4.14 <u>TS 3.8.2.2, DC [Direct Current] Sources Shutdown</u>

The licensee proposed to delete the action to suspend movement of irradiated fuel, if the LCO is not met. The licensee states in the LAR that the reason for this deletion is that, "[t]he Fuel Handling Accident analysis no longer credits the mitigation systems that are dependent upon this source of electrical power." The TS 3.8.2.2 is currently applicable in "MODES 5 and 6."

The NRC staff reviewed the licensee's AST DBA analyses for FHA. The accident mitigation capabilities of the following SSCs:

1. FHB Exhaust Air System and its actuation instrumentation;

2. Control Room Makeup and Cleanup Filtration System and its actuation instrumentation; and

3. Containment Ventilation Isolation RCB Purge Radioactivity – High actuation instrumentation (i.e., on RCB Purge isolation)

are no longer credited in the licensee's DBA analysis.

For shutdown Modes 5 and 6, it follows that the relevant portions of the DC electrical power subsystem used in support of these SSCs' mitigation capabilities are no longer required during movement of irradiated fuel. Therefore, for Modes 5 and 6 of TS 3.8.2.2 during movement of irradiated fuel, the relevant portions of the DC electrical power subsystem used in support of these SSCs are no longer part of a primary success path used to mitigate a DBA that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. For the relevant portions of the DC electrical power subsystem used in support of these SSCs, the criteria for inclusion in TS, pursuant to 10 CFR 50.36(d)(2)(ii), are no longer met.

Based on the above, the NRC staff finds the proposed change acceptable.

3.4.15 TS 3.8.3.2, Onsite Power Distribution Shutdown

The licensee proposes to delete the action to suspend movement of irradiated fuel if the LCO is not met.

The licensee stated in its LAR that the reason for the change is that the FHA analysis no longer credits the mitigation systems that are dependent upon these sources of electrical power.

The NRC staff reviewed the licensee's request as follows. The TS 3.8.3.2 is currently applicable to MODES 5 and 5. The NRC staff reviewed the licensee's AST DBA analyses for FHA. The accident mitigation capabilities of the following SSCs:

1. FHB Exhaust Air System and its actuation instrumentation;

2. Control Room Makeup and Cleanup Filtration System and its actuation instrumentation; and

3. Containment Ventilation Isolation RCB Purge Radioactivity – High actuation instrumentation (i.e. on RCB Purge isolation)

are no longer credited in the design-basis FHA analysis.

For shutdown Modes 5 and 6, it follows that the relevant portions of AC, DC, and AC vital bus electrical power distribution subsystems used to support these SSCs' mitigation capabilities are no longer required during movement of irradiated fuel.

For Modes 5 and 6 of TS 3.8.3.2 during movement of irradiated fuel, the relevant portions of the AC, DC, and AC vital bus electrical power distribution subsystems used in support of these SSCs are no longer part of a primary success path used to mitigate a DBA that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. For the relevant portions of the AC, DC, and AC vital bus electrical power distribution subsystems used in support of these SSCs, the criteria for TS inclusion per 10 CFR 50.36(d)(2)(ii) are no longer met.

Based on the above, the staff finds the proposed change acceptable.

3.4.16 TS 3/4.9.4, Containment Building Penetrations during Refueling Operations

The licensee proposed to delete TS 3/4.9.4 in its entirety.

The licensee stated in the LAR that the reason for the change is that containment isolation is no longer credited in AST DBA analysis during core alterations or movement of irradiated fuel.

The TS 3.9.4 currently applies “[d]uring CORE ALTERATIONS or movement of irradiated fuel within the containment.”

The NRC staff reviewed the proposed change. In Section 4.4, “Fuel Handling Accident,” of the LAR the licensee concluded that of the accidents postulated to occur during core alterations (i.e., inadvertent criticality, FHA and the loading of a fuel assembly or control component in the wrong location) only the FHA results in cladding damage and potential radiological release.

The analysis of Section 4.4 was performed assuming at least 42 hours had elapsed since the fuel was last critical. The proposed amendment takes credit for the normal decay of irradiated fuel rather than crediting certain active mitigation systems. Containment isolation was not a requirement in the analysis. Neither FHB nor RCB filtration was credited in the analysis. The FHA Dose Results of Table 4.4.3 of the AST analysis for the EAB, LPZ, CR, and TSC indicate that the doses at each receptor (rem TEDE) are substantially below regulatory limits set forth in 10 CFR 50.67 and the acceptance criteria set forth in RG 1.183.

Consistent with the guidance contained in Revision 2 of TSTF-51, Attachment 5 of the LAR contains the necessary commitments (Commitment #s 3 and 6) to isolate the RCB for postulated FHA in the RCB and further reduce dose by natural decay and to enable the ventilation system to draw the release from a postulated FHA in the containment in the proper direction such that it can be monitored (not treated). These commitments will further reduce dose below those “Dose (0-2 hour)” values listed in Table 4.4.3 of the AST analysis. The licensee states that it plans to meet the above necessary commitments, and the licensee plans to include changes in the STP, Units 1 and 2 TRM.

The reactor sub-criticality requirements contained in TS 3/4.9.4 are no longer valid. These requirements can be deleted since sub-criticality requirements are now located in the STP, Units 1 and 2 TRM. The licensee’s LAR states:

> The decay time specification was relocated from TS to the TRM with approval of STPNOC licensing amendments 145 and 133 to Units 1 and 2, respectively. In the safety evaluation for this TS change, the NRC staff found that the "decay time” requirement specification does not need to be in the TS because it is not needed to ensure the decay time limit is met. This is because certain operational steps, such as containment entry, pressure vessel head removal, and cavity flood-up must be completed before fuel movement in the vessel is possible following critical operation. These preliminary activities require more than 42 hours to complete. The NRC staff found that these physical limitations are adequate to ensure compliance with the 42-hour limit (relocated to the TRM). Thus, including the decay time limit in TSs is not needed to ensure this limit is met.

For TS 3/4.9.4 the LCOs, Actions, and SRs for the containment building penetrations are no longer required during core alterations or during movement of irradiated fuel within the containment. The containment building penetrations of TS 3/4.9.4 are no longer parts of a primary success path that actuate to mitigate a DBA that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Accordingly, these containment building penetrations no longer satisfy the criteria for inclusion in the TS per 10 CFR 50.36(d)(2)(ii) during core alterations or during movement of irradiated fuel within the containment.

Based on the above, the staff finds the proposed deletion of TS 3/4.9.4 acceptable.

3.4.17 TS 3/4.9.9, Containment Ventilation Isolation during Refueling Operations

The licensee proposed to delete TS 3/4.9.9 in its entirety. The licensee states in the LAR that the reason for the change is that containment isolation is no longer credited in AST DBA analysis during core alterations or movement of irradiated fuel.

The Applicability for TS 3.9.9 reads, "[d]uring CORE ALTERATIONS or movement of irradiated fuel within the containment." The NRC staff reviewed the proposed change. In Section 4.4, "Fuel Handling Accident," of the LAR, the licensee concluded that of the accidents postulated to occur during core alterations (i.e., inadvertent criticality, FHA, and the loading of a fuel assembly or control component in the wrong location) only the FHA results in cladding damage and potential radiological release.

The analysis of Section 4.4 was performed assuming at least 42 hours had elapsed since the fuel was last critical. The proposed amendment takes credit for the normal decay of irradiated fuel rather than crediting certain active mitigation systems. Containment isolation was not a requirement in the analysis. Neither FHB nor RCB filtration was credited in the analysis. The FHA Dose Results of Table 4.4.3 of the AST analysis for the EAB, LPZ, CR, and TSC indicate that the doses at each receptor (rem TEDE) are substantially below the reference values of 10 CFR 50.67 and the regulatory guidance in RG 1.183.

Consistent with the guidance contained in Revision 2 of TSTF-51, Attachment 5 of the LAR contains the necessary commitments (Commitment #s 3 and 6) to isolate the RCB for postulated FHA in the RCB, to further reduce dose by natural decay and to enable the ventilation system to draw the release from a postulated FHA in the containment in the proper direction such that it can be monitored (not treated). These commitments will further reduce dose below those "Dose (0-2 hour)" values listed in Table 4.4.3 of the AST analysis.

The operability of Containment Ventilation Isolation System for closure of each of the purge and exhaust penetrations is no longer required during core alterations or during movement of irradiated fuel within the containment for the reasons indicated above. During these activities, the Containment Ventilation Isolation System is no longer part of a primary success path that actuates to mitigate a DBA that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Accordingly, during core alterations or during movement of

irradiated fuel within the containment, the Containment Ventilation Isolation System no longer satisfies the criteria for inclusion in the TS pursuant to 10 CFR 50.36(d)(2)(ii).

Based on the above, the staff finds the deletion of TS 3/4.9.9 is acceptable.

3.4.18 TS 3/4.9.12, Fuel Handling Building Exhaust Air System during Refueling Operations

The licensee proposes to delete TS 3/4.9.12 in its entirety. The licensee stated that the reason for the proposed change is, that the accident mitigation capabilities of the FHB HVAC system are no longer credited in AST DBA analysis. The applicability for TS 3.9.12 reads, "[w]henever irradiated fuel is in the spent fuel pool."

The NRC staff reviewed the proposed change. The staff reviewed the licensee's AST DBA analyses for LOCA, FHA, MSLB, SGTR, CREA, and LRA. The accident mitigation capabilities of the FHB HVAC filter system are no longer credited in any of these design bases accident analyses.

In summary, the staff finds the proposed deletion of TS 3.9.12 is acceptable.

Whenever irradiated fuel is being moved or when conducting crane operation with loads over the spent fuel pool the licensee commits in Attachment 5 of the LAR:

1. to keep as OPERABLE or to restore at least one train of the Fuel Handling Building Exhaust Air to an OPERABLE status within two hours;

2. to keep radiation monitoring instrumentation functional to ensure that a release following a Fuel Handling Accident is monitored; and

3. to place at least one train of the FHB Exhaust Air system into operation within two hours of a Fuel Handling Accident.

This is consistent with the guidance contained in TSTF-51, Revision 2. The purpose of these commitments is to maintain the FHB Ventilation System and the associated radiation monitoring available to reduce doses even further below that provided by natural decay and to avoid unmonitored releases; and to enable the FHB Ventilation System to draw the release from a postulated FHA in the FHB in the proper direction such that it can be treated and monitored.

To meet the above commitments, the licensee plans to include the changes to STP, Unit 1 and 2 TRM, and keep the FHB Exhaust Air System in a near operable status (i.e., "restore to Operable within two hours") and to keep the radiation monitoring instrumentation functional during the movement of irradiated fuel or when conducting crane operation with loads over the spent fuel pool.

The FHB Exhaust Air System is no longer part of a primary success path that actuates to mitigate a DBA that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Accordingly, for all Modes of plant operation, the FHB Exhaust Air System no longer satisfies the criteria for inclusion in the TS per 10 CFR 50.36(d)(2)(ii).

3.4.19 Conclusion

The staff finds that the proposed TS changes satisfy the relevant regulatory requirements and guidance (e.g., GDC, regulations, RGs, NUREGs, etc.) of Section 2.0. Therefore, the NRC staff finds that the TS changes proposed by the licensee are acceptable.

3.5 Commitments

The purpose of the following licensee commitments is to maintain the FHB Ventilation System and associated radiation monitoring available to reduce doses even further below those provided by the natural decay and to avoid unmonitored releases; and to enable the FHB Ventilation System to draw the release from a postulated FHA in the FHB in the proper direction such that it can be treated and monitored. The implementation of these commitments will provide additional defense in depth in the event of an FHA, however, the licensee has shown that these commitments are not required in order to meet the applicable dose criteria. Therefore, the following commitments are found to be acceptable to the staff.

3.5.1 FHB Exhaust Air Operability

Whenever fuel is being moved in the spent fuel pool or when conducting crane operation with loads over the spent fuel pool, at least one train of FHB Exhaust Air shall be OPERABLE or capable of being restored to an OPERABLE status within 2 hours.

3.5.2 Control Room Makeup and Cleanup Filtration System Operability

Whenever irradiated fuel is being moved or when conducting crane operation with loads over the spent fuel pool, at least one Control Room Makeup and Cleanup Filtration System shall be OPERABLE or capable of being restored to an OPERABLE status within 2 hours.

3.5.3 RCB Closure Conditions During Irradiated Fuel Movement

Whenever irradiated fuel is being moved within the RCB, the following will be closed or capable of being closed within 2 hours.

1. The equipment hatch,

2. At least one door in the Auxiliary Airlock and one door in the Personnel Airlock, and

3. All other penetrations providing direct access from the containment atmosphere to the outside atmosphere.

3.5.4 FHB Exhaust Air Operation

Within 2 hours of an FHA in the FHB, at least one train of FHB Exhaust Air will be placed in operation.

3.5.5 Control Room Makeup and Cleanup Filtration System Operation

Within 2 hours of an FHA, at least one Control Room Makeup and Cleanup Filtration System will be placed in operation.

3.5.6 Post FHA RCB Closure Actions

Within 2 hours of an FHA in the RCB, the following actions will be taken:

1. Close the equipment hatch,

2. Close at least one of the Auxiliary Airlock doors and one of the Personnel Airlock doors, and

3. Close all other penetrations providing direct access from the containment atmosphere to the outside atmosphere.

3.5.7 SFP Radiation Monitoring Functionality

Whenever irradiated fuel is being moved or when conducting crane operation with loads over the spent fuel pool, radiation monitoring instrumentation will remain functional to ensure that a release following an FHA is monitored.

The following commitment addresses a licensee identified condition related to this amendment request. A letter dated December 13, 2006, Westinghouse NSAL-06-15, advised operators of Westinghouse plants that the single-failure scenario for the SGTR analysis that licensees used in their accident analysis may not be limiting. The licensee has evaluated the applicability of NSAL-06-15 against the accident analysis assumptions and has determined that the current single-failure assumption for the STP SGTR analysis is not limiting. Therefore, the licensee is operating under compensatory measures to meet regulatory dose limits. The licensee plans to resolve this condition at the earliest opportunity so that the assumptions, including the limiting single failure, for the SGTR accident analysis described herein are consistent with the plant response to this event.

3.6 Administrative Limit for RCS DEI

Until a plant modification is completed for supporting the limiting single-failure assumptions in the SGTR analysis, STP will maintain an administrative limit for RCS DEI so that the radiological dose guidelines for the SGTR analysis remain bounding and compliance with GDC 19 is continued. The staff's acceptance of this position is discussed in detail in the SGTR portion of this SE and is based on the licensee's demonstration that with the administrative limit in place the CR dose consequence from an SGTR accident will be well within the dose acceptance criterion.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use or surveillance of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 published in the *Federal Register* on July 31, 2007 (72 FR 41788). 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 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.

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Table 1
STP, Units 1 and 2 Radiological Consequences Expressed as TEDE [1]
(rem)

Design Basis Accidents	EAB [2]	LPZ [3]	CR
Loss-of-coolant accident	5.6E+00	2.8E+00	3.7E+00
Main steamline break accident [4]	5.2E-02	4.1E-02	1.1E-01
Steam Generator Tube Rupture [4]	2.4E+00	9.2E-01	2.2E+00
Dose acceptance criteria	2.5E+01	2.5E+01	5.0E+00
Control Rod Ejection Accident [6]	8.6E-01	1.7E+00	2.4E+00
Control Rod Ejection Accident [7]	5.5E-01	2.0E-01	4.1E-01
Control Rod Ejection Accident Total	1.4E+00	1.9E+00	2.8E+00
Fuel Handling A	8.3E-01	3.0E-01	3.4E+00
Dose acceptance criteria	6.3E+00	6.3E+00	5.0E+00
Main steamline break accident [5]	8.5E-01	6.6E-01	1.7E+00
Steam Generator Tube Rupture [5]	1.1E+00	4.4E-01	1.0E+00
Locked Rotor Accident	1.9E+00	1.5E+00	3.9E+00
Dose acceptance criteria	2.5E+00	2.5E+00	5.0E+00

[1] Total effective dose equivalent
[2] Exclusion area boundary
[3] Low population zone
[4] Pre-accident iodine spike
[5] Concurrent iodine spike
[6] Assumes containment release
[7] Assumes secondary-side release

Note: Licensee results are expressed to two significant figures

Table 2
STP, Units 1 and 2 CR and TSC Atmospheric Dispersion Factors

Source Location / Duration		χ/Q (sec/m3)
For evaluating releases from containment leakage for the LOCA and CREA		
	0 - 2 hours	2.17E-04
	2 - 8 hours	1.37E-04
	8 - 24 hours	6.15E-05
	24 - 96 hours	4.14E-05
	96 - 720 hours	2.30E-05
For evaluating releases from the plant vent for the LOCA and FHA		
	0 - 2 hours	7.12E-04
	2 - 8 hours	5.28E-04
	8 - 24 hours	2.04E-04
	24 - 96 hours	1.61E-04
	96 - 720 hours	9.76E-05
For evaluating releases from the PORV for the MSLB, SGTR, and LRA		
	0 - 2 hours	6.13E-04
	2 - 8 hours	3.27E-04
	8 - 24 hours	1.55E-04
	24 - 96 hours	1.01E-04
	96 - 720 hours	7.18E-05

Table 3
Offsite Atmospheric Dispersion Factors (sec/m3)

Receptor / Duration		χ/Q (sec/m3)
EAB		
	0 - 2 hours	1.44E-04
	2 - 8 hours	N/A
	8 - 24 hours	N/A
	24 - 96 hours	N/A
	96 - 720 hours	N/A
LPZ		
	0 - 2 hours	5.27E-05
	2 - 8 hours	2.24E-05
	8 - 24 hours	1.46E-05
	24 - 96 hours	5.75E-06
	96 - 720 hours	1.51E-06

Table 4
STP, Units 1 and 2 Control Room Data and Assumptions and Direct shine results

Control room envelope calculated volume	304,000 ft^3
Control room envelope net free air volume	274,080 ft^3
CR filtered make up flow in dose model	0 cfm (not credited)
CR emergency mode filtered recirculation flow rate	8600 cfm
CR pressurization (make up) filter efficiencies	
Elemental iodine	0%
Organic iodine	0%
Particulates	0%
CR emergency mode credited recirculation filter efficiency	
Elemental iodine	95%
Organic iodine	95%
Particulates	99%
CR emergency mode unfiltered in-leakage	
Unfiltered in-leakage	100 cfm
Makeup flow rate assumed to be unfiltered	2200
Total modeled unfiltered in-leakage	2300 cfm
CR operator breathing rate	
0 - 720 hours	3.5E−04 m3/sec
CR occupancy factors	
0 - 24 hours	1.0
24 - 96 hours	0.6
96 - 720 hours	0.4
LOCA CR gamma shine component doses (rem)	
RCB Activity	0.101
External Cloud	0.014
CR makeup filters	Negligible
CR recirculation filters	0.004
Electrical penetration room	0.017

Table 5 (Page 1 of 2)
STP, Units 1 and 2 Data and Assumptions for the LOCA

Power level for radiological source term	4100 MWt	
Power level for RCS steam release	3876 MWt	
Activity in coolant blowdown	1% clad defect	
Coolant blowdown mass	9.3E+05 lbm	
Volumetric flow rate due to open purge valves	142,000 cfm	
Duration of flow through open purge valves	23 seconds	
Concurrent LOOP	Yes	
Primary containment volumetric leak rate		
0 - 24 hours	0.3 %/day	
24 - 720 hours	0.15 %/day	
Primary containment volume sprayed region	2.7E+06 ft^3	
Primary containment volume unsprayed region	6.8E+05 ft^3	
Flow rate between sprayed and unsprayed regions	152,475 cfm	
Elemental iodine spray removal coefficients	Sprayed region	Unsprayed region
0.0 - 0.039 hours	4.5 hr -1	4.5 hr -1
0.039 – 1.855 hours	24.5 hr -1	4.5 hr -1
1.855 - 720 hours	0.0 hr -1	0.0 hr -1
Particulate spray removal coefficients	Sprayed region	Unsprayed region
0.0 - 0.039 hours	0.0 hr -1	0.0 hr -1
0.039 – 2.185 hours	6.9 hr -1	0.0 hr -1
2.185 - 720 hours	0.7 hr -1	0.0 hr -1
Iodine chemical form in containment atmosphere		
cesium iodide	95%	
elemental iodine	4.85%	
organic iodine	0.15%	
Volume of water in containment sump	61,486 ft^3	
Assumed ESF leakage (two times limit of 4140 cc/hr)	8280cc/hr	
Assumed ECCS leakage start time	T = 0	
Time dependant ECCS Flashing fractions		
Time in hours	Sump pH	Flashing fraction
0 to 24	7	10%
24 to 480	6.9	16%
480 to 720	6.8	25%

Table 5 (Page 2 of 2)
STP, Units 1 and 2 Data and Assumptions for the LOCA

Time dependant ECCS Flashing fractions

Time in hours	Sump pH	Flashing fraction
0 to 24	7	10%
24 to 480	6.9	16%
480 to 720	6.8	25%

Chemical form of released iodine from ECCS leakage

Elemental	97%
Organic	3%
Particulate	0%

Containment sump pH	6.8
Containment sump DF used in dose analysis	60
Volume of electrical penetration area	101,477 ft^3
Containment electrical penetration leak rate	100 sccm[1] per penetration
Number of containment electrical penetrations	18
Ventilation exhaust rate for electrical penetration area	833 cfm

[1] Standard cubic centimeters per minute

Table 6
STP, Units 1 and 2 Data and Assumptions for the FHA

Core thermal power level	4100 MWt
Minimum post-shutdown fuel handling time (decay time)	42 hours
Core radial peaking factor	1.7
Fuel clad damage gap release fractions	
I-131	8%
Remainder of halogens	5%
Kr-85	10%
Remainder of noble gases	5%
Number of fuel assemblies in the core	193
Total number of fuel rods per assembly	264
Total number of rods in the core	50,952
Number of rods assumed to fail in the FHA	
Dropped assembly	264
Impacted assembly	50
Total number of failed rods	314
Minimum pool water depth	23 feet
Pool DF	
Noble gases and organic iodine	1
Aerosols	Infinite
Elemental iodine (23 ft of water cover)	285
Overall iodine (23 ft of water cover)	200 (effective DF)
Chemical form of iodine in pool	
Elemental	99.85%
Organic	0.15%
Duration of release to the environment	Instantaneous release
Release location	Plant vent
FHB ventilation filter efficiencies (not credited)	0
CR ventilation filter efficiencies (not credited)	0

Table 7
STP, Units 1 and 2 Data and Assumptions for the MSLB Accident

Power level for radiological source term	4100 MWt
Power level for RCS steam release	3876 MWt
RCS density	8.33 lbm/gallon
RCS mass	2.658E+08 gm
Initial RCS equilibrium activity	1.0 µCi/gm DEI, 1% FF
Maximum pre-accident spike iodine concentration	60 µCi/gm DEI
Steam flow rate	1.574E+07 lbm/hour
SG Node volume	
Intact SG	5.937E+04 gallons
Faulted SG	1.979E+04 gallons
Primary-to-secondary leak rates	
Intact SG	0.65 gpm
Faulted SG	0.35 gpm
Time to establish RHR system terminating steam release	8 hours
Duration of leakage through MSIV above-seat orifices	36 hours
Release from faulted SG	
Instantaneous release	214,000 lbm
Subsequent release from feedwater systems	385,000 lbm
Total	599,000 lbm
Steaming release from the three intact SGs	
0-2 hours	452,000 lbm
2-8 hours	1,080,000 lbm
Release from above MSIV seat drains	
Intact SGs	5.79 lbm/sec
Faulted SG	1.93 lbm/sec
Concurrent spike appearance rate multiplier	500
Secondary coolant iodine activity	0.1 µCi/gm DEI
Iodine species released from flashed RCS primary-to-secondary leakage to environment	
Elemental	4.85 %
Organic	0.15 %
Particulate	95.0 %
SG secondary side iodine partition coefficients	
Elemental	100
Organic	1(none)
Particulate	100
Resulting iodine species released from the secondary side	
Elemental	4.2 %
Organic	13.1 %
Particulate	82.7 %

Table 8 (Page 1 of 2)
STP, Units 1 and 2 Data and Assumptions for the SGTR Accident

Power level for radiological source term	4100 MWt
Power level for RCS steam release	3876 MWt
RCS and secondary coolant system density	8.33 lbm/gallon
RCS mass	2.685E+08 gm
Initial RCS equilibrium activity	1.0 µCi/gm DEI, 1% Clad Defect
Maximum pre-accident spike iodine concentration	60 µCi/gm DEI
SG Node volume	
Intact SGs (ISGs)	5.937E+04 gallons
Ruptured SG (RSG)	1.979E+04 gallons
Secondary coolant system mass	659,412 lbm
Initial secondary side equilibrium activity	0.1 µCi/gm DEI
Primary-to-secondary leakage rate	
ISGs	0.65 gpm
RSG	0.35 gpm
Credited operator action times	
Diagnose the SGTR event	T= 10 minutes post-SGTR
Close PORV Block Valve (BV) on RSG	T= 25 minutes post-SGTR
Release from above MSIV Seat Drains	
ISGs	5.79 lbm/sec
RSG	1.93 lbm/sec
Steam Flow Rate	1.574E+07 lbm/hr
Effective DF for releases from condenser prior to LOOP	10,000

Conservative SGTR mass release in lbm during time period in seconds used for Dose Analysis

			Steam Release to Atmosphere		
			Ruptured SG		Intact SGs
Event @ Initial Time	Time Period (sec)	Break flow	Flashed	Total	Total
SGTR	0:66.5	5517	846	117,600	336,000
Rx Trip LOOP	66.5:607	35,515	2374	14,092	51,919
RSG isolated	607:1507	70,466	9660	183,911	10,317
RSG PORV BV closed	1507:2087	41,613	3448	0	187,936
Flashing in RSG ends	2087:5248	138,562	0	0	385,021
Break flow terminated	5248:7380	0	0	0	327,062
Dose model changes	7380:28,980	0	0	70,056	1,621,851
RHR Entry	28,980:129,800	0	0	0	0
End of orifice release	129,800				

Table 8 (Page 2 of 2)
STP, Units 1 and 2 Data and Assumptions for the SGTR Accident

RCS Iodine Inventory (Ci) for 8-hr concurrent spike with an appearance rate factor of 335	
I-131	1.16E+05
I-132	3.77E+05
I-133	2.16E+05
I-134	1.57E+05
I-135	7.44E+05
Iodine species for flashed break RCS flow to the environment	
Elemental	4.85 %
Organic	0.15 %
Particulate	95.0 %
SG secondary side iodine partition coefficients	
Elemental	100
Organic	1(none)
Particulate	100
Resulting iodine species released from the secondary side	
Elemental	4.2 %
Organic	13.1 %
Particulate	82.7 %

Table 9
STP, Units 1 and 2 Data and Assumptions for the CRE Accident

Power level for radiological source term	4100 MWt
Power level for RCS steam release	3876 MWt
RCS and secondary coolant system density	8.33 lbm/gallon
RCS mass	2.658E+08 gm
SG mass	659,412 lbm
Initial RCS activity concentration	60 µCi/gm DEI, 1% Clad Defect
Initial secondary side activity concentration	0.1 µCi/gm DEI, 1% Clad Defect
Primary-to-secondary side leakage	1 gpm
Fuel centerline melt	0.25% of core
Fuel clad damage	10% of core
Minimum time to release initial SG mass	191 seconds
Steam flow rate for release of initial SG mass	2.07E+05 lbm/min
Max time for primary-to-secondary pressure equilibrium	4500 seconds
Containment volume	
For radionuclide dilution	3.38E+06 ft^3
For leakage rate	3.41E+06 ft^3
Containment leak rate	
0 to 24 hours	0.3%/day (by volume)
24 - 720 hours	0.15%/day (by volume)
SG secondary side iodine partition coefficients	
Elemental	100
Organic	1(none)
Particulate	100

Chemical form of released iodine	To containment	From SGs
Particulate	95%	82.7%
Elemental	4.85%	4.2%
Organic	0.15%	13.1%

Steam flow rate	1.574E+07 lbm/hour

Control rod ejection steam release (lbm) for time period (hrs)

Event	Time interval	PORV	Above seat drains	Total
CREA	0:1.25	15,535,885	34,740	15,570,625
PORV release ends	1.25:36	0	965,772	965,772
Orifice release ends	36:720	0	0	0
	Total	15,535,885	1,000,512	16,536,397

-Table 10
STP, Units 1 and 2 Data and Assumptions for the Locked Rotor Accident

Power level for radiological source term	4100 MWt
Power level for RCS steam release	3876 MWt
Initial RCS activity concentration	60 µCi/gm DEI, 1% Clad Defect
Initial secondary side activity concentration	0.1 µCi/gm DEI, 1% Clad Defect
Fuel clad damage	10% of core
RCS and secondary coolant system density	8.33 lbm/gallon
RCS mass	2.658E+08 gm
SG mass	659,412 lbm
Primary-to-secondary leakage	
SGs without tube recovery	0.65 gpm
SGs with tube recovery	0.35 gpm
Release from above (MSIV) seat drains	
SGs without tube recovery	5.79 lbm/sec
SGs with tube recovery	1.93 lbm/sec
Steam flow rate	1.574E+07 lbm/hour
SG secondary side iodine partition coefficients	
Elemental	100
Organic	1(none)
Particulate	100
Chemical form of iodine released from the SGs	
Particulate	82.7%
Elemental	4.2%
Organic	13.1%

Locked rotor accident steam release (lbm) for time period (hrs)

Event	Time interval	PORV	Above seat drains	Total
LRA	0:2	640,000	55,584	695,584
	2:8	1,120,000	166,752	1,286,752
PORV release ends	8:12	0	111,168	111,168
	12:36	0	667,008	667,008
Orifice release ends	36:720	0	0	0
	Total	1,760,000	1,000,512	2,760,512