Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

July 31, 2002

TVA-BFN-TS-405 10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Washington, D. C. 20555

Gentlemen:

In the Matter of	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
		50-296

BROWNS FERRY NUCLEAR PLANT (BFN) - UNITS 1, 2, AND 3 - LICENSE AMENDMENT - ALTERNATIVE SOURCE TERM

In accordance with the provisions of 10 CFR 50.4 and 10 CFR 50.90, TVA is submitting a request for an amendment to licenses DPR-33, DPR-52 and DPR-68 that supports a full scope application of an Alternative Source Term (AST) methodology for BFN Units 1, 2, and 3. Specifically, TVA requests revision to the licensing and design basis to reflect the application of AST methodology on Units 1, 2, and 3 and approval of associated Technical Specifications (TSs) changes which are justified by the AST analyses. TVA is also proposing deletion of a completed License Condition to licenses DPR-52 and DPR-68.

On December 23, 1999, the NRC published 10 CFR 50.67, "Accident Source Term." This regulation provides a mechanism for licensed power reactors to replace the traditional source term used in design basis accident analyses with an AST. 10 CFR 50.67 requires licensees who seek to revise their current accident source term in design basis radiological consequence analyses to apply for a license amendment under 10 CFR 50.90.

Full Scope AST analyses were performed following the guidance in Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and Standard Review Plan Section 15.0.1,

A001

"Radiological Consequences Analyses using Alternative Source Terms." AST analyses were performed for the four Updated Final Safety Analysis Report (UFSAR) Chapter 14 BFN Design Basis Accidents (DBA) that could potentially result in control room and offsite doses. These include the Loss of Coolant Accident (LOCA), the Main Steam Line Break Accident, the Refueling Accident, and the Control Rod Drop Accident. The analyses demonstrated that using AST methodologies, post-accident control room and offsite doses remain within the regulatory limits.

TVA proposes implementation of this change through both revisions to the TS and UFSAR. Proposed changes in the licensing basis for BFN resulting from AST application include the following:

- TS and UFSAR changes that reflect revised design requirements regarding the use of the Standby Liquid Control (SLC) System to buffer the suppression pool preventing iodine re-evolution following a postulated design basis LOCA.
- TS revisions to reflect the relaxation of Secondary Containment, Standby Gas Treatment, and Control Room Emergency Ventilation System requirements. AST analyses do not take credit for secondary containment during the movement of irradiated fuel and during core alterations. Therefore, these system TS may be made less restrictive.
- TS revisions to remove the requirements to test the charcoal filters for Standby Gas Treatment and Control Room Emergency Ventilation Systems. AST analyses does not take credit for adsorption of elemental iodine, organic iodine, or noble gases by the charcoal. Therefore, the charcoal filters are no longer required and the associated TS may be deleted. Additionally, the testing requirements are being revised to add limits for pressure drop without charcoal adsorbers.

TVA is also requesting deletion of Facility Operating License Condition 2.C.(4) for Units 2 and 3. The license condition required that TVA perform analyses of the design bases LOCA, confirm compliance with off-site and on-site dose limits, obtain NRC approval of the results, and make any needed modifications. These actions are complete and, therefore, this License Condition is no longer applicable.

Since the three units share a common refueling floor, the completed AST radiological dose analysis for the refueling accident is valid for all of the units. Unit 1 is currently shutdown, defueled, and in long term layup. The AST analyses for the remaining three DBAs have been performed for Units 2 and 3, but not for Unit 1. As required by existing Unit 1 License Condition 2.C.(4), TVA will verify that the required AST analyses needed for the remaining DBAs for Unit 1 are complete, and submit them for NRC review and approval prior to Unit 1 restart. Because the three units are essentially identical, TVA expects that the Unit 1 analyses will show comparable results as Units 2 and 3. Therefore, TVA is requesting this amendment and TS change be approved for Unit 1.

In support of a project to uprate the licensed thermal power of BFN Units 2 and 3, TVA determined that it was appropriate to adopt AST. This decision was communicated to the NRC staff in a meeting in Rockville, Maryland on December 5, 2001. Additional meetings were held on January 16, 2002, and July 10, 2002, between TVA and the staff to discuss the specifics of TVA's planned AST submittal, including the incorporation of Unit 1 TS changes. In those meetings, the analysis approach, submittal content, and schedule were discussed.

The current operating license allows Units 2 and 3 to operate at a maximum power level of 3458 megawatts thermal (MWt). TVA is currently engaged in an Extended Power Uprate (EPU) project to increase the maximum licensed thermal power for Units 2 and 3 to 3952 MWt. Therefore, the AST analyses which have been performed considered the core isotopic values for the current and future vendor products at EPU conditions and this license amendment is based on a bounding core isotopic inventory.

The use of AST changes the analytical treatment of the DBA radiological consequences. The use of AST has no direct impact on the probability of the evaluated DBAs. The changes in implementing AST methodology and the other changes requested by this license amendment do not increase the core damage frequency or the large early release frequency. Therefore, this TS change request is not being submitted as a "risk-informed licensing action" as defined by Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific changes to the Licensing Basis."

Several other Boiling Water Reactors (Duane Arnold, Brunswick, Grand Gulf, Hope Creek, Clinton, and Perry) have previously provided justification for the use of AST utilizing a similar approach. These applications have been approved by NRC.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the change is exempt from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). The BFN Plant Operations Review Committee and the Nuclear Safety Review Board have reviewed this proposed change and determined that operation of BFN Units 1, 2, and 3 in accordance with the proposed change will not endanger the health and safety of the public. Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and enclosures to the Alabama State Department of Public Health.

Enclosure 1 to this letter provides the description and justification of the proposed change. This includes TVA's determination that the proposed change does not involve a significant hazards consideration and is exempt from environmental review. Enclosure 2 contains copies of the appropriate marked-up TS pages from Units 1, 2, and 3 to show the proposed TS changes. UFSAR Section 3.8 is being revised to describe the new safety function of the SLC System Enclosure 3 provides marked up to indicate the proposed license change. Enclosure 4 provides the BFN Alternative Source Term Safety Assessment.

RG 1.183 recommends that changes to the UFSAR that reflect the revised analyses be submitted to the staff. Enclosure 5 provides proposed changes to UFSAR Section 14.6 that have been identified as requiring revision to reflect the AST analyses. Enclosure 5 also provides a matrix identifying other sections in the UFSAR that are currently under evaluation for change. The final UFSAR changes will be completed as required by BFN procedures following approval of this amendment request.

TVA requests the approval of the proposed license amendment for Units 1, 2, and 3 by April of 2003 and requests that the revised TS be made effective within 60 days of NRC approval. There are no new regulatory commitments associated with this submittal. If you have any questions about this change, please contact me at (256) 729-2636.

Pursuant to 28 U.S.C. § 1746 (1994), I declare under penalty of perjury that the foregoing is true and correct.

Executed on this 31st day of July, 2002.

Sincerely,

T. E. Abney
Manager of Licensing
and Industry Affairs

Enclosures
cc: See page 6

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TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

PROPOSED LICENSE AMENDMENT
ALTERNATIVE SOURCE TERM
INDEX OF ENCLOSURES

Enclosure 1 - DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGES

Enclosure 2 - MARKED PAGES - Technical Specifications

Enclosure 3 - Marked pages - UFSAR

Enclosure 4 - Alternative Source Term Safety Assessment

Enclosure 5 - UFSAR Chapter 14.6 Mark-Ups

ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

PROPOSED LICENSE AMENDMENT
ALTERNATIVE SOURCE TERM
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

I. DESCRIPTION OF THE PROPOSED CHANGE

In accordance with the provisions of 10 CFR 50.4, 10 CFR 50.67, and 10 CFR 50.90, TVA is requesting an amendment to licenses DPR-33, DPR-52, and DPR-68 to implement an Alternative Source Term (AST) for BFN Units 1, 2, and 3. This change request includes revisions to the licensing and design basis to reflect the full application of AST methodology and changes to the Technical Specifications (TS) justified by the AST analyses.

This full implementation of AST analyses will modify the licensing bases by adopting AST methodology which replaces the current accident source term with an alternative source term as prescribed in 10 CFR 50.67 and establishes the 10 CFR 50.67 total effective dose equivalent (TEDE) dose limits as a new acceptance criteria.

Since all three units share a common refueling floor the completed AST radiological dose analysis for the refueling accident is valid for all three Units. Unit 1 is currently shutdown, defueled, and in long term layup. The AST analyses for the remaining Design Bases Accidents have been performed for Units 2 and 3, but not for Unit 1 As required by Unit 1 Facility License 2.C.(4), TVA will verify that the required analyses needed for the remaining DBAs for Unit 1 are complete, and submit them for NRC review and approval prior to Unit 1 restart. Because the three units are essentially identical, it is expected that the Unit 1 analyses will show comparable results to Units 2 and 3. Therefore, TVA is requesting this amendment and TS change be approved for Unit 1.

The current operating license allows Units 2 and 3 to operate at a maximum power level of 3458 megawatts thermal (MWt). TVA is currently engaged in an Extended Power Uprate (EPU) project to increase the maximum licensed thermal power for Units 2 and 3 to 3952 MWt. Therefore, the AST analysis which have been performed considered the core isotopic values at EPU conditions and this license amendment is based on a bounding core isotopic inventory.

This enclosure provides the description and justification of the proposed changes. This includes TVA's determination that the proposed change does not involve a significant hazards consideration and is exempt from environmental review. Enclosure 2 contains copies of the appropriate marked-up pages from Units 1, 2, and 3 TS which show the proposed changes. UFSAR Section 3.8 is being revised to describe the new safety function of the SLC System. This UFSAR section is included in Enclosure 3 marked up to indicate the proposed licensing bases changes. Enclosure 4 provides the BFN AST Safety Assessment. This enclosure provides a summary description and basis for the acceptability of the proposed changes associated with the AST methodology.

RG 1.183 recommends that changes to the UFSAR that reflect the revised analyses be submitted to the staff. Enclosure 5 provides changes to UFSAR Section 14.6 that have been identified requiring revision to reflect the AST analyses. A matrix identifying other sections in the UFSAR that are currently under evaluation for change is also provided in Enclosure 5. The final UFSAR changes will be completed as required by BFN procedures following approval of this amendment request.

The license amendment revises BFN Units 1, 2, and 3 TS and the UFSAR to implement the AST analysis. The revisions are as follows:

Technical Specification Changes

- TS 3.1.7, Standby Liquid Control (SLC) System, is being changed to revise the required amount of sodium pentaborate from Ü3007 gallons to Ü4000 gallons. Additionally, a new surveillance requirement to verify that the sodium pentaborate concentration is Ü8.0% by weight is being added. SLC system operability will also be required in Mode 3. These changes implement AST methodology regarding the use of SLC to buffer the suppression pool following a Loss of Coolant Accident involving fuel damage.

- TS Table 3.3.6.2-1, Secondary Containment Isolation Instrumentation, is being revised to delete the requirement for operable secondary containment instrumentation during core alterations and movement of irradiated fuel assemblies in the secondary containment. The AST analyses dose not take credit for the secondary containment function. Removal of this requirement is further justified by the AST analyses.

- TS Table 3.3.7.1-1, Control Room Emergency Ventilation (CREV) System Instrumentation, is being revised to delete the requirement for operable CREV instrumentation during core alterations and movement of irradiated fuel assemblies in the secondary containment. The AST analyses does not take credit for automatic CREV initiation during core alterations. Removal of these requirements is further justified by the AST analyses.

- TS 3.6.4.1, Secondary Containment, TS 3.6.4.2, Secondary Containment Isolation Valves (SCIVs), TS 3.6.4.3, Standby Gas Treatment (SGT) System and TS 3.7.3, CREV System is being revised to delete the requirement for operability during core alterations and movement of irradiated fuel assemblies in the secondary containment. The AST analyses does not take credit for these functions. Removal of these requirements is further justified by the AST analyses.

- TS 5.5.7, Ventilation Filter Testing Program (VFTP), is being revised to delete sections b and c which require testing of charcoal adsorbers in the SGT and CREV Systems, respectively. Since AST analyses takes no credit for charcoal filters, the testing requirements are being removed from the TS. Additionally, TS 5.5.7 Section d is being revised to add limits for pressure drop testing without charcoal adsorbers (and after-filters) installed. BFN has no specific plans for physical removal of these adsorbers; however, removal would not require further license amendment.

UFSAR Changes

UFSAR Section 3.8, Standby Liquid Control System, is being revised to describe the new safety function of maintaining the suppression pool water pH at or above 7.0 to prevent iodine re-evolution following a LOCA that involves fuel damage.

Administrative Change Deletion of License Condition

Facility Operating License Condition 2.C.(4) for Unit 2 and Unit 3 is being deleted. This license condition required TVA to perform analyses of the design basis LOCA, confirm compliance with off-site and on-site dose limits, obtain NRC approval, and make any needed modifications. Since these requirements have been completed, this license condition is no longer applicable.

II. REASON FOR THE PROPOSED CHANGE

Approval of this change will provide a more realistic source term for BFN that will result in a more accurate assessment of DBA radiological doses. This allows relaxation of some current licensing basis requirements. Adopting the AST may also support future evaluations and license amendments.

III. BACKGROUND

On December 23, 1999, the NRC published 10 CFR 50.67, "Accident Source Term." This regulation provides a mechanism for licensed power reactors to replace the current accident source term used in DBA analyses with an alternative source term. The direction provided in 10 CFR 50.67 is that licensees who seek to revise their current accident source term in design basis radiological consequences analyses apply for a license amendment under 10 CFR 50.90.

In July 2000, NRC published Regulatory Guide 1.183, "Alternative Source Terms For Evaluating Design Basis Accidents at Nuclear Power Reactors." Regulatory

Guide (RG) 1.183 provides guidance to licensees on acceptable applications of alternative source terms; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. Since then, several BWRs (Duane Arnold, Brunswick, Grand Gulf, Hope Creek, Clinton, and Perry) have submitted license amendments to adopt AST. These amendments have been approved by NRC. TVA reviewed these submittals, including the associated NRC requests for additional information and Safety Evaluations for inclusion into this submittal.

AST analyses for DBAs were performed following the guidance in RG 1.183 and Standard Review Plan (SRP) 15.0.1, "Radiological Consequences Analyses Using Alternative Source Term." Acceptance criteria consistent with that required by 10 CFR 50.67 were used to replace the current design basis source term acceptance criteria. The AST analyses were performed for four BFN DBAs that could potentially result in control room and offsite doses. These include the LOCA, the Main Steam Line Break Accident, the Refueling Accident, and the Control Rod Drop Accident.

Browns Ferry is a three unit site. Units 2 and 3 are in operation, each having a licensed thermal power of 3458 MW. Unit 1 has a licensed thermal power of 3293 MW. Unit 1 is shutdown, defueled and in long-term layup. Activities are currently underway for restart of Unit 1 within 5 years. Each of these units is a General Electric BWR-4 boiling water reactor with a Mark I containment design. The three units share a common refueling floor, and the three control rooms are all located in a single habitability zone. A 600 foot tall offgas stack serves all three units. Browns Ferry Units 2 and 3 have previously implemented modifications that make the main steam lines seismically rugged. This established an alternative leakage treatment leakage path using the main steam system piping and the main condenser for post accident dose mitigation for main steam isolation valve leakage.

In support of a project to uprate the licensed thermal power of BFN Units 2 and 3, TVA determined that it was appropriate to adopt AST. This decision was communicated to the NRC staff in a meeting in Rockville, Maryland on December 5, 2001. Additional meetings were held on January 16, 2002, and July 10, 2002, between TVA and the staff to discuss the specifics of TVA's planned AST submittal, including the incorporation of Unit 1 TS changes. In those meetings, the analysis approach, submittal content, and schedule were discussed.

The following provides a background discussion on control room habitability as requested by the NRC during the January 2002 meeting.

Control Room Habitability Discussion

In a July 31, 1992 letter (Reference 1), TVA described corrective actions to resolve self-identified deficiencies in the design of the CREV System. These corrective actions were related to on the discovery that there was substantial unfiltered inleakage into the control room.

In October of 1997, TVA requested a license amendment to allow BFN to operate Units 2 and 3 at an uprated power level of 3458 megawatts thermal.

In reviewing the license amendment for power uprate, NRC requested additional information regarding TVA's unfiltered inleakage into the control room. In a May 7, 1998 letter (Reference 2), NRC requested that TVA include the effects of MSIV leakage to the turbine building with regard to control room dose, exclusion area boundary (EAB) dose and low population zone (LPZ) dose. In addition, NRC requested an assessment of control room dose, EAB dose, and LPZ dose due to leakage from Emergency Core Cooling Systems (ECCS) consistent with NRC SRP 15.6.5, Appendix B.

In a September 8,1998 letter (Reference 3), NRC issued a license amendment to allow operation of BFN Units 2 and 3 at 3458 megawatts thermal power. As part of the amendment, TVA concurred and NRC added a license condition that required performance of an analysis of the DBA LOCA to confirm compliance with General Design Criteria (GDC) 19 and offsite limits considering MSIV leakage and ECCS leakage and submit the results by March 31, 1999. The results of this analysis were transmitted to the NRC in a letter dated March 30, 1999 (Reference 4). This letter stated that the calculated doses were bounded by the allowable doses prescribed by 10 CFR 50 Appendix A, GDC 19 and 10 CFR 100 with the unfiltered control room inleakage.

On August 3, 1999 (Reference 5), NRC provided a Safety Evaluation (SE) acknowledging the revised dose calculation to be the analyses of record for the radiological consequence for a Design Bases Accident (DBA) LOCA.

TVA has used this NRC approved dose analysis, including the unfiltered control room inleakage, to support another license amendment. By application dated September 28, 1999 (Reference 6), supplemented February 4, 2000 (Reference 7), TVA requested a revision to the Units 2 and 3 TS to increase the allowable leakage for the main steam line isolation valves. By letter dated March 14, 2000 (Reference 8), NRC approved these TS amendments.

In the March 14, 2000, SE the staff concluded that there was reasonable assurance that the BFN control room will be habitable during a postulated DBA. This is based on (1) the relative magnitude of the infiltration currently assumed in the BFN analysis (3717 cfm of which is unfiltered), (2) the site X/Q values, (3) actions previously taken by TVA, and (4) the low probability of a design basis event occurring that could result in radioactivity releases sufficient to challenge the ability of control room personnel to protect the health and safety of the public.

In summary, TVA conducted tests and determined the unfiltered inleakage into the control room. This inleakage has been included in the BFN licensing basis and has been accepted by NRC.

IV. Safety Evaluation

A. Alternative Source Term

BFN has performed a full scope analysis of the AST as defined in RG 1.183. A detailed description of AST analysis is provided in Enclosure 4 and the methods and results of the analysis are summarized in this section. The analysis included the following:

1. Identification of the core source term based on plant specific analysis of core fission product inventory.
2. Determination of the release fractions for the four BFN DBAs that could potentially result in control room and offsite doses. These are the LOCA, the main steam line break accident, the refueling accident, and the control rod drop accident.
3. Calculation of fission product deposition rates and removal efficiencies.
4. Calculation of offsite and control room personnel TEDE.
5. Evaluation of suppression pool pH requirements to ensure that the particulate iodine deposited into the suppression pool does not re-evolve and become airborne as elemental iodine.
6. Calculation of a new control room atmospheric dispersion factor (X/Q) for a main steam line break accident instantaneous ground level puff release.
7. Evaluation of other related design and licensing bases such as NUREG-0737, "Clarification of TMI Action Plan Requirements."

The radiological dose analyses for AST have been performed assuming reactor operation at Extended Power Uprate conditions (3952 Mwt). This results in a conservative estimate of fission product releases for current licensed power of the units. BFN Units 2 and 3 currently have a maximum licensed thermal power of 3458 Mwt. However, TVA is actively engaged in an EPU project to increase reactor power to 3952 MWt.

AST Methodology

Implementation of AST included the following:

1. Development of a bounding plant-specific core fission product inventory.

2. Introduction of a new X/Q for an instantaneous ground level puff release to the atmosphere for the main steam line break accident.

3. No credit is taken for CREV or SGT System charcoal adsorption for any DBA.

4. No credit is taken for CREV or SGT System HEPA filter particulate removal for any DBA except LOCA.

5. New requirements for post-LOCA SLC System operation for suppression pool pH control along with calculation of sodium pentaborate (SPB) quantity requirements were developed.

The AST analyses were performed in accordance with RG 1.183. The results were evaluated to confirm compliance with the acceptance criteria presented in 10 CFR 50.67 and General Design Criteria 19 of 10 CFR 50, Appendix A.

Evaluation

DBA accident analyses documented in Chapter 14 of the BFN UFSAR that potentially result in control room and offsite doses were addressed using methods and input assumptions consistent with the AST methodology. The following BFN DBAs were addressed:

- Loss of Coolant Accident (LOCA), UFSAR Section 14.6.3
- Main Steam Line Break Accident, UFSAR Section 14.6.5
- Refueling Accident, UFSAR Section 14.6.4
- Control Rod Drop Accident, UFSAR Section 14.6.2

The AST control room dose analyses are applicable for all three unit control rooms. The Unit 1 and 2 control rooms are shared in a common room with Unit 1 at one end and Unit 2 at the other. The Unit 3 control room, though separated from the Unit 1 and 2 control room, is part of the same control bay habitability zone. The refueling accident radiological consequence analysis is applicable to all three units since the refuel zone is common.

Results

LOCA

The radiological consequences of the DBA LOCA were analyzed. The post-accident doses are the result of the following activity considerations:

1. Primary to secondary containment leakage. This leakage is directly released into secondary containment and filtered by SGT System prior to elevated release through the plant stack with stack bypass released at

ground level. No credit is taken for SGT or CREV System charcoal adsorber action.

2. ECCS leakage into the secondary containment. This leakage is directly released into the secondary containment environment and the airborne portion is filtered by SGT System prior to elevated release through the plant stack with stack bypass fraction released at ground level. No credit is taken for SFG or CREV System charcoal adsorber.

3. MSIV leakage from the primary containment into the main condenser (with a fraction that bypasses the main condenser directly to the atmosphere). Leakage passes through the alternate MSIV leakage pathway to the main condenser with credit for deposition before it is released, undiluted and unfiltered, through the turbine building vents.

4. Harden Wet Well Vent leakage from primary containment. This leakage is directly released (after a eight hour delay) to an elevated release through the plant stack.

5. Post-DBA LOCA radiation shine dose to personnel within the control room from activity released to the reactor building and from activity contained in Core Spray System piping.

Loss Of Coolant Accident

For the AST LOCA analysis, Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and control room calculated doses remain within the regulatory limits. These results are summarized in the following table along with results for the LOCA analysis using the current source term.

LOCA Radiological Consequence Analysis (rem TEDE)			
	Offsite Dose		Control Room Dose
Dose Component	**EAB**	**LPZ**	
Base of Stack	—	1.08E-2	4.49E-3
Top of Stack	—	5.68E-1	2.43E-1
Turbine Building Roof	—	3.02E-1	1.13E-1
ECCS Leakage - Base of Stack	—	1.25E-2	1.21E-2
ECCS Leakage - Top of Stack	—	3.52E-1	1.12E-1
Shine	—	N/A	7.62E-1
TOTAL	**1.02**	**1.25**	**1.25**
Regulatory Limit	25	25	5
Current Analysis (Regulatory Limit) - rem	1.67E-01 (25) Gamma 1.01E-01 (300) Beta 5.84 (300) Thyroid	4.82E-01 (25) Gamma 4.84E-01 (300) Beta 8.6 (300) Thyroid	6.83E-01 (5) Gamma 1.58E-01 (30) Beta 2.95E+01 (30) Thyroid

Main Steam Line Break Accident

For the Main Steam Line Break analysis EAB, LPZ, and control room calculated doses remain within the regulatory limits for the two cases analyzed. The control room doses were determined using the new X/Q value for an instantaneous ground level puff release. These results are summarized in the table below along with the results from the current source term analysis.

Main Steam Line Break Accident Radiological Consequence Analysis (rem TEDE)			
	Offsite Dose		Control Room Dose
Case	EAB	LPZ	
3.2 μCi/gm DE I-131	1.30E-1	6.52E-2	4.09E-2
32 μCi/gm DE I-131	1.30	6.52E-1	4.09E-1
Regulatory Limit	25	25	5
Current Analysis (Regulatory Limit) - rem[1]	3.72E-01 (25) Gamma 1.56E-01 (300) Beta 2.99E+01 (300) Thyroid	1.86E-01 (25) Gamma 7.80E-02 (300) Beta 1.49E+01 (300) Thyroid	5.30E-02 (5) Gamma 3.27E-02 (30) Beta 1.05E+01 (30) Thyroid

[1] Current analysis are based on 32 μCi/gm DE I-131 limit.

Refueling Accident

For the AST design basis refueling accident the EAB, LPZ, and control room calculated doses are within the regulatory limits. The results are summarized in the table below along with the results of the current source term analyses.

Refueling Accident Radiological Consequence Analysis (rem TEDE)			
	Offsite Dose		Control Room Dose
Case	EAB	LPZ	
24 Hours after shutdown	6.7E-01	3.3E-01	3.8E-01
Regulatory Limit	6.30	6.30	5
Current Analysis (Regulatory Limit) - rem	3.37E-01 (25) Gamma 5.77E-01 (300) Beta 3.32E+01 (300) Thyroid	1.68E-01 (25) Gamma 2.89E-01 (300) Beta 1.66E+01 (300) Thyroid	4.94E-02 (5) Gamma 4.96E-01 (30) Beta 1.74 (30) Thyroid

Control Rod Drop Accident

The radiological consequences of the design basis control rod drop accident using AST methodology were analyzed. The EAB, LPZ, and control room calculated doses remain within the regulatory limits after AST implementation. The results are summarized in the table below along with the results of the current source term analyses.

Control Rod Drop Accident Radiological Consequence Analysis (rem TEDE)			
Case	Offsite Dose EAB	Offsite Dose LPZ	Control Room Dose
Power Operation	1.19	6.82E-01	2.48E-01
Regulatory Limit	6.30	6.30	5
Current Analysis (Regulatory Limit) - rem	1.52 (25) Gamma 1.07 (300) Beta 1.58E+01 (300) Thyroid	8.58E-01 (25) Gamma 6.04E-01 (300) Beta 1.58E+01 (300) Thyroid	3.86E-02 (5) Gamma 4.32E-01 (30) Beta 6.3 (30) Thyroid

Suppression Pool pH Control

The AST LOCA analysis takes credit for minimization of re-evolution of elemental iodine from the suppression pool, which is strongly dependent on suppression pool pH. The analysis assumed that sodium pentaborate SPB was injected via SLC within several hours of the onset of a LOCA. The conservative modeling of the primary containment cabling results in the production of a large amount of hydrochloric acid. Using the assumptions of a minimum of 4000 gallons of ≥8% by weight injectable SPB solution, the minimum suppression pool pH at 30 days post-LOCA remains above 7.0. This pH satisfies the conditions for inhibiting the release of the chemical form of elemental iodine from the containment. This quantity of SLC is above current TS SR 3.1.7 requirements of 3007 gallons. Therefore, TS revisions are proposed which increase the quantity of SLC required to be maintained as shown in Enclosure 2.

Based on the AST analysis for suppression pool pH control, the SLC system will also be credited for limiting radiological dose following a design basis recirculation pipe break LOCAs involving fuel damage. However, the SLC system will not be re-classified as a safety system, but will retain the current classification as described in UFSAR Section 3.8.

Main Steam Line Break Accident Puff Release Dispersion Factor

In support of the AST Main Steam Line Break analysis, a new control room X/Q value for an instantaneous ground level puff release to the atmosphere was calculated for use in the radiological dose analysis. This X/Q value is shown in the table below.

Main Steam Line Break Accident Instantaneous Ground Level Puff Release X/Q Value (Main Steam Line Break Accident Only)	
Time Period	Control Room (sec/m^3)
46 secs	4.60E-4

NUREG-0737 Evaluation

The revised analyses includes consideration of the impacts of AST methodology for several NUREG-0737 items. These are summarized below.

- Post-Accident Vital Area Access and Sampling - The results of the revised post-accident mission dose calculations demonstrate that the current calculated doses (based on TID-14844 source terms) bound the doses that would be calculated based on AST source terms. The evaluated mission doses remain less than 5 rem TEDE (NUREG-0737, Items II.B.2 and II.B.3).

- Post-Accident Radiation Monitor - The containment high range radiation monitors used to monitor post-accident primary containment radiation levels were evaluated for the impact of AST. The monitors continue to provide their design function and envelope the projected radiation rates. (NUREG-0737, Item II.F.1).

- Control Room Radiation Protection - The resultant doses to the control room for each of the four DBAs analyzed for AST have been determined. In each case the control room dose is less than 5 rem TEDE (NUREG-0737 items III.A.1.2 and III.D.3.4).

- Radioactive Sources Outside the Primary Containment - The contribution of radiological dose consequences as a result of radiation shine and ECCS leakage was determined as part of the radiological dose analysis for the LOCA and found acceptable (NUREG-0737, Item III.D.1.1).

Conclusion

Radiological dose analyses were performed using AST methodology for the four BFN DBAs with a potential for control room and offsite doses. Control room and offsite doses remain within regulatory requirements.

B. Pressure Drop Testing of ESF Ventilation System

TS 5.5.7.d addresses the pressure drop test across the combined HEPA filters, prefilters, and charcoal adsorbers for the CREVS and SGT systems. As discussed earlier, AST radiological analyses do not take credit for charcoal filters in the CREVS and SGT Systems. Although BFN has no specific plans for the physical removal of these adsorbers, TS 5.5.7.d must be revised to include the case in which the charcoal adsorbers and associated after-filters may be removed. The after-filters are present to capture any charcoal fines and have not been credited for any radioactivity removal.

A plant modification to remove these filters would result in a decrease in the pressure drop through the filter trains for these systems. Accordingly, the new TS limits for the pressure drop tests have been decreased to reflect the potential removal of the charcoal adsorber and after-filter. The revised limits will ensure that appropriate testing criteria exists for the potential removal of the charcoal adsorber and resulting system modification effects.

V. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

TVA is submitting a request for amendment to the Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3 Technical Specifications (TS). The proposed amendment is a full implementation of an alternative source term (AST) for the Units 1, 2, and 3 operating licenses, adopting AST methodology by revising the current accident source term and replacing it with an accident source term as prescribed in 10 CFR 50.67.

AST analyses were performed using the guidance provided by Regulatory Guide 1.183, "Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000, and Standard Review Plan Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms." The four limiting design basis accidents (DBAs) considered were the Control Rod Drop Accident, the Refueling Accident, the Loss of Coolant Accident, and the Main Steam Line Break Accident.

TVA has concluded that operation of BFN Units 1, 2, and 3 in accordance with the proposed change to the TS does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation in accordance with 10 CFR 50.91(a)(1) of the three standards set forth in 10 CFR 50.92(c).

A. **<u>The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>**

The AST and those plant systems affected by implementing AST do not initiate DBAs. The AST does not affect the design or operation of the facility; rather, once the occurrence of an accident has been postulated, the new source term is an input to evaluate the consequences. The implementation of the AST has been evaluated in the analyses for the limiting DBAs at BFN.

The equipment affected by the proposed change is mitigative in nature and relied upon following an accident. The proposed changes to the TS do revise certain performance requirements. However, these changes will not involve a revision to the parameters or conditions that could contribute to the initiation of a design basis accident discussed in Chapter 14 of the BFN Updated Final Safety Analysis Report.

Plant specific radiological analyses have been performed and, based on the results of these analyses, it has been demonstrated that the dose consequences of the limiting events considered in the analyses are within the regulatory guidance provided by the NRC for use with the AST. This guidance is presented in 10 CFR 50.67, Regulatory Guide 1.183, and Standard Review Plan Section 15.0.1. Therefore, the proposed amendment does not result in a significant increase in the consequences or a significant increase the probability of any previously evaluated accident.

B. **<u>The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.</u>**

Implementation of AST does not alter any design basis accident initiators. These changes do not affect the design function or mode of operations of systems, structures, or components in the facility prior to a postulated accident. Since systems, structures, and components are operated essentially no differently after the AST implementation, no new failure modes are created by this proposed change. Therefore, the proposed license amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated.

C. **<u>The proposed amendment does not involve a significant reduction in a margin of safety.</u>**

The changes proposed are associated with a revision to the licensing basis for BFN. The results of accident analyses revised in support of the proposed change are subject to the acceptance criteria in 10 CFR 50.67. The analyzed events have been carefully selected, and the analyses supporting this submittal have been performed using approved methodologies. The dose consequences of these limiting events are within the acceptance criteria provided by the regulatory guidance as presented in 10 CFR 50.67, Regulatory Guide 1.183, and SRP 15.0.1.

Therefore, because the proposed changes continue to result in dose consequences within the applicable regulatory limits, the changes are considered to not result in a significant reduction in a margin of safety.

VI. ENVIRONMENTAL IMPACT CONSIDERATION

The proposed change does not involve a significant hazards consideration, a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

References:

1. TVA letter to NRC dated July 31, 1992, Browns Ferry Nuclear Plant - Resolution of Control Room Emergency Ventilation System (CREV) Issues.

2. NRC letter to TVA dated May 7, 1998, Browns Ferry Nuclear Plant, Units 2 and 3: Request for Additional Information Relating To Technical Specification Change No. TS-384 - Power Uprate Operation.

3. NRC Letter to NRC dated September 8, 1998, Issuance to Amendments Re: Power Uprate - Browns Ferry Plant, Units 2 and 3.

4. TVA Letter to NRC dated March 30, 1999, Browns Ferry Nuclear plant (BFN) - Resolution Of Control Room Emergency Ventilation (CREV) System Issues With Regard To License Condition Associated With Units 2 and 3 power Uprate Operating License Amendments 254 and 214.

5. NRC Letter to TVA dated August 3, 1999, Safety Evaluation Supplement, Browns Ferry Nuclear Plant Units 2 and 3 - Radiological Dose Calculations Associated With Power Uprate License Amendment Nos. 254 and 214.

6. TVA Letter to NRC dated September 28, 1999, Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Technical Specification (TS) Change 399 - Increased Main Steam Isolation Valve (MSIV) Leakage Rate Limits and Exemption From 10 CFR 50 Appendix J.

7. TVA Letter to NRC dated February 4, 2000, Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Technical Specifications Change 399 - Increased Main Steam isolation Valve (MSIV) Leakage Rate Limits and Exemption From 10 CFR 50 Appendix J.

8. NRC Letter to TVA dated March 14, 2000, Browns Ferry Nuclear Plant, Units 2 and 3 - Issuance of Amendments Regarding Limits on Main Steam Isolation Valve Leakage (TAC Nos. MA6405 and MA6406).

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

PROPOSED LICENSE AMENDMENT
ALTERNATIVE SOURCE TERM
MARKED PAGES - TECHNICAL SPECIFICATIONS

AFFECTED PAGE LIST

The following pages have been revised. On the affected pages the revised portions have been highlighted. A line has been drawn through the deleted text and a double underline for new or revised text.

Operating License

Unit 2	Unit 3
Page 4	Page 4

Technical Specifications

Unit 1	Unit 2	Unit 3
3.1-23	3.1-23	3.1-23
3.1-24	3.1-24	3.1-24
3.1-25	3.1-25	3.1-25
3.1-26	3.1-26	3.1-26
3.3-64	3.3-65	3.3-65
3.3-69	3.3-70	3.3-70
3.6-44	3.6-44	3.6-44
3.6-45	3.6-45	3.6-45
3.6-47	3.6-47	3.6-47
3.6-49	3.6-49	3.6-49
3.6-51	3.6-51	3.6-51
3.6-52	3.6-52	3.6-52
3.6-53	3.6-53	3.6-53
3.7-8	3.7-9	3.7-9
3.7-9	3.7-10	3.7-10
3.7-10	3.7-11	3.7-11
5.0-15	5.0-15	5.0-15
5.0-16	5.0-16	5.0-16

(3) The licensee is authorized to relocate certain requirements included in Appendix A and the former Appendix B to licensee-controlled documents. Implementation of this amendment shall include the relocation of these requirements to the appropriate documents, as described in the licensee's application dated September 6, 1996, as supplemented May 1, August 14, November 5 and 14, December 3, 4, 11, 22, 23, 29, and 30, 1997, January 23, March 12, April 16, 20 and 28, May 7, 14, 19, and 27, and June 2, 5, 10 and 19, 1998, evaluated in the NRC staff's Safety Evaluation enclosed with this amendment. This amendment is effective immediately and shall be implemented within 90 days of the date of this amendment.

(4) ~~TVA will perform an analysis of the design basis loss-of-coolant accident to confirm compliance with General Design Criterion (GDC) 19 and offsite limits considering main steam isolation valve leakage and emergency core cooling system leakage. The results of this analysis will be submitted to the NRC for its review and approval by March 31, 1999. Following NRC approval, any required modifications will be implemented during the refueling outages scheduled for Spring 2000 for Unit 3 and Spring 2001 for Unit 2. TVA will maintain the ability to monitor radiological conditions during emergencies and administer potassium iodide to control room operators to maintain doses within GDC 19 guidelines. This ability will be maintained until the required modifications, if any, are complete. This amendment is effective immediately~~.

(5) Classroom and simulator training on all power uprate related changes that affect operator performance will be conducted prior to operating at uprated conditions. Simulator changes that are consistent with power uprate conditions will be made and simulator fidelity will be validated in accordance with ANSI/ANS 3.5-1985. Training and the plant simulator will be modified, as necessary, to incorporate changes identified during startup testing. This amendment is effective immediately.

(5)(a) Deleted

(6) Deleted.

(7) Deleted.

(3) The licensee is authorized to relocate certain requirements included in Appendix A and the former Appendix B to licensee-controlled documents. Implementation of this amendment shall include the relocation of these requirements to the appropriate documents, as described in the licensee's application dated September 6, 1996, as supplemented May 1, August 14, November 5 and 14, December 3, 4, 11, 22, 23, 29, and 30, 1997, January 23, March 12, April 16, 20, and 28, May 7, 14, 19, and 27, and June 2, 5, 10 and 19, 1998, evaluated in the NRC staff's Safety Evaluation enclosed with this amendment. This amendment is effective immediately and shall be implemented within 90 days of the date of this amendment.

(4) ~~TVA will perform an analysis of the design basis loss-of-coolant accident to confirm compliance with General Design Criterion (GDC) 19 and offsite limits considering main steam isolation valve leakage and emergency core cooling system leakage. The results of this analysis will be submitted to the NRC for its review and approval by March 31, 1999. Following NRC approval, any required modifications will be implemented during the refueling outages scheduled for Spring 2000 for Unit 3 and Spring 2001 for Unit 2. TVA will maintain the ability to monitor radiological conditions during emergencies and administer potassium-iodide to control room operators to maintain doses within GDC-19 guidelines. This ability will be maintained until the required modifications, if any, are complete. This amendment is effective immediately~~.

(5) Classroom and simulator training on all power uprate related changes that affect operator performance will be conducted prior to operating at uprated conditions. Simulator changes that are consistent with power uprate conditions will be made and simulator fidelity will be validated in accordance with ANSI/ANS 3.5-1985. Training and the plant simulator will be modified, as necessary, to incorporate changes identified during startup testing. This amendment is effective immediately.

(6) The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Browns Ferry Physical Security Plan", with revisions submitted through May 24, 1988; "Browns Ferry Security Personnel Training and Qualification Plan", with revisions submitted through April 16, 1987; and "Browns Ferry Safeguards Contingency Plan", with revisions submitted through June 27, 1986. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, ~~and~~ 2, and 3.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify available volume of sodium pentaborate solution (SPB) is ≥ ~~3007~~ 4000 gallons.	24 hours
SR 3.1.7.2	Verify continuity of explosive charge.	31 days
SR 3.1.7.3	Verify the SPB concentration is ≥ 8.0% by weight	31 days AND Once within 24 hours after water or boron is added to solution
SR 3.1.7~~.3~~ .4	Verify the SPB concentration is ≤ 9.2% by weight.	31 days AND Once within 24 hours after water or boron is added to solution
	OR	
	Verify the concentration and temperature of boron in solution are within the limits of Figure 3.1.7-1.	Once within 8 hours after discovery that SPB concentration is > 9.2% by weight AND 12 hours

		thereafter
SR 3.1.7.~~4~~ 5	Verify the minimum quantity of Boron-10 in the SLC solution tank and available for injection is ≥ 186 pounds.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.~~5~~6	Verify the SLC conditions satisfy the following equation: $$\frac{(\ C\ )(\ Q\ )(\ E\ )}{(13\text{ wt. }\%)(86\text{ gpm})(19.8\text{ atom}\%)} \geq 1$$ where, C = sodium pentaborate solution concentration (weight percent) Q = pump flow rate (gpm) E = Boron-10 enrichment (atom percent Boron-10)	31 days AND Once within 24 hours after water or boron is added to the solution
SR 3.1.7.~~6~~7	Verify each pump develops a flow rate ≥ 39 gpm at a discharge pressure ≥ 1275 psig.	18 months
SR 3.1.7.~~7~~8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	18 months on a STAGGERED TEST BASIS
SR 3.1.7.~~8~~9	Verify all piping between storage tank and pump suction is unblocked.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.~~9~~10	Verify sodium pentaborate enrichment is within the limits established by SR 3.1.7.~~5~~6 by calculating within 24 hours and verifying by analysis within 30 days.	18 months AND After addition to SLC tank
SR 3.1.7.~~10~~11	Verify each SLC subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position, or can be aligned to the correct position.	31 days

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low, Level 3	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≥ 538 inches above vessel zero
2. Drywell Pressure - High	1,2,3	2	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 2.5 psig
3. Reactor Zone Exhaust Radiation - High	1,2,3, (a)~~(b)~~	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 100 mR/hr
4. Refueling Floor Exhaust Radiation - High	1,2,3, (a)~~(b)~~	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 100 mR/hr

(a) During operations with a potential for draining the reactor vessel.

~~(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in secondary containment.~~

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Ventilation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low, Level 3	1,2,3,(a)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≥ 538 inches above vessel zero
2. Drywell Pressure - High	1,2,3	2	B	SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 2.5 psig
3. Reactor Zone Exhaust Radiation - High	1,2,3 (a)~~,(b)~~	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 100 mR/hr
4. Refueling Floor Exhaust Radiation - High	1,2,3, (a)~~,(b)~~	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 100 mR/hr
5. Control Room Air Supply Duct Radiation - High	1,2,3, (a)~~,(b)~~	1	D	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 270 cpm above background

(a) During operations with a potential for draining the reactor vessel.

~~(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in the secondary containment.~~

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment,~~
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Secondary containment inoperable ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	~~C.1 ----------NOTE---------- LCO 3.0.3 is not applicable. ------------------------------~~	
	~~Suspend movement of irradiated fuel assemblies in the secondary containment.~~	~~Immediately~~
	~~AND~~	
	~~C.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	C.~~3~~ 1 Initiate action to suspend OPDRVs.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

LCO 3.6.4.2 Each SCIV shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment,~~
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

--NOTES--

1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIVs.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more penetration flow paths with one SCIV inoperable.	A.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	8 hours
	<u>AND</u>		
			(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	~~D.1~~	~~-----NOTE-----~~ ~~LCO 3.0.3 is not applicable.~~ ~~----------~~ ~~Suspend movement of irradiated fuel assemblies in the secondary containment.~~	~~Immediately~~
	~~AND~~		
	~~D.2~~	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~		
	D.~~3~~ 1	Initiate action to suspend OPDRVs.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Three SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment,~~
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One SGT subsystem inoperable.	A.1	Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Be in MODE 4.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	~~NOTE~~ ~~LCO 3.0.3 is not applicable.~~	
	C.1 Place two OPERABLE SGT subsystems in operation.	Immediately
	OR	
	~~C.2.1 Suspend movement of irradiated fuel assemblies in secondary containment.~~	~~Immediately~~
	~~AND~~	
	~~C.2.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	C.2~~.3~~ Initiate action to suspend OPDRVs.	Immediately
D. Two or three SGT subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two or three SGT subsystems inoperable ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	~~E.1 ----------NOTE---------- LCO 3.0.3 is not applicable. ------------------------------ Suspend movement of irradiated fuel assemblies in secondary containment.~~	~~Immediately~~
	~~AND~~	
	~~E.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	E.~~3~~ 1 Initiate action to suspend OPDRVs.	Immediately

3.7 PLANT SYSTEMS

3.7.3 Control Room Emergency Ventilation (CREV) System

LCO 3.7.3 Two CREV subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment,~~
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREV subsystem inoperable.	A.1 Restore CREV subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	C.1 ~~NOTE~~ ~~LCO 3.0.3 is not applicable.~~ Place OPERABLE CREV subsystem in pressurization mode.	Immediately
	OR	
	~~C.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.~~	~~Immediately~~
	~~AND~~	
	~~C.2.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	C.2~~.3~~ Initiate action to suspend OPDRVs.	Immediately
D. Two CREV subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two CREV subsystems inoperable ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	~~E.1 ------NOTE------ LCO 3.0.3 is not applicable. ---------------------~~	
	~~Suspend movement of irradiated fuel assemblies in the secondary containment.~~	~~Immediately~~
	~~AND~~	
	~~E.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	E.~~3~~ 1 Initiate action to suspend OPDRVs.	Immediately

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

b. ~~Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ 1.0% when tested in accordance with ANSI N510-1975 at the system flowrate specified below, ± 10%.~~

~~ESF Ventilation System~~	~~Flowrate (cfm)~~
~~SGT System~~	~~9000~~
~~CREV System~~	~~3000~~

~~This testing shall be performed 1) every 24 months, 2) after partial or complete replacement of the charcoal adsorber bank, 3) after any structural maintenance on the system housing, or 4) following significant painting, fire, or chemical release in any ventilation zone communicating with the system.~~

~~c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, shows a methyl iodide efficiency ≥ 90% when tested in accordance with ASTM D3803-1989.~~

~~This testing shall be performed 1) every 24 months, 2) after every 720 hours of system operation, or 3) following significant painting, fire, or chemical release in any ventilation zone communicating with the system.~~

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

~~d~~ b. Once every 24 months demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below at the system flowrate specified below, ± 10%:

ESF Ventilation System	Delta P with charcoal adsorbers installed (inches water)	Delta P without charcoal adsorbers installed (inches water)	Flowrate (cfm)
SGT System	7	5	9000
CREV System	6	4	3000

~~e~~ c. Once every 24 months demonstrate that the heaters for the SGT System dissipate ≥ 40 kW when tested in accordance with ANSI N510-1975.

5.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained downstream of the offgas recombiners, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

a. The limits for concentrations of hydrogen downstream of the offgas recombiners and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

(continued)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, ~~and~~ 2, and 3.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify available volume of sodium pentaborate solution (SPB) is ≥ ~~3007~~ 4000 gallons.	24 hours
SR 3.1.7.2	Verify continuity of explosive charge.	31 days
SR 3.1.7.3	Verify the SPB concentration is ≥ 8.0% by weight	31 days AND Once within 24 hours after water or boron is added to solution
SR 3.1.7~~3~~ 4	Verify the SPB concentration is ≤ 9.2% by weight.	31 days AND Once within 24 hours after water or boron is added to solution
	OR	
	Verify the concentration and temperature of boron in solution are within the limits of Figure 3.1.7-1.	Once within 8 hours after discovery that SPB concentration is > 9.2% by weight AND

		12 hours thereafter
SR 3.1.7~~.4~~.5	Verify the minimum quantity of Boron-10 in the SLC solution tank and available for injection is ≥ 186 pounds.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.1.7.~~5~~6	Verify the SLC conditions satisfy the following equation: $$\frac{(\ C\ )(\ Q\ )(\ E\ )}{(13\text{ wt. }\%)(86\text{ gpm})(19.8\text{ atom}\%)} \geq 1$$ where, C = sodium pentaborate solution concentration (weight percent) Q = pump flow rate (gpm) E = Boron-10 enrichment (atom percent Boron-10)	31 days AND Once within 24 hours after water or boron is added to the solution
SR 3.1.7.~~6~~7	Verify each pump develops a flow rate ≥ 39 gpm at a discharge pressure ≥ 1325 psig.	24 months
SR 3.1.7.~~7~~8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 months on a STAGGERED TEST BASIS
SR 3.1.7.~~8~~9	Verify all piping between storage tank and pump suction is unblocked.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.~~9~~10	Verify sodium pentaborate enrichment is within the limits established by SR 3.1.7.~~5~~6 by calculating within 24 hours and verifying by analysis within 30 days.	24 months AND After addition to SLC tank
SR 3.1.7.~~10~~11	Verify each SLC subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position, or can be aligned to the correct position.	31 days

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low, Level 3	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≥ 528 inches above vessel zero
2. Drywell Pressure - High	1,2,3	2	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 2.5 psig
3. Reactor Zone Exhaust Radiation - High	1,2,3, (a)~~(b)~~	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 100 mR/hr
4. Refueling Floor Exhaust Radiation - High	1,2,3, (a)~~(b)~~	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 100 mR/hr

(a) During operations with a potential for draining the reactor vessel.

~~(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in secondary containment.~~

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Ventilation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low, Level 3	1,2,3,(a)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≥ 528 inches above vessel zero
2. Drywell Pressure - High	1,2,3	2	B	SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 2.5 psig
3. Reactor Zone Exhaust Radiation - High	1,2,3 (a)~~,(b)~~	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 100 mR/hr
4. Refueling Floor Exhaust Radiation - High	1,2,3, (a)~~,(b)~~	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 100 mR/hr
5. Control Room Air Supply Duct Radiation - High	1,2,3, (a)~~,(b)~~	1	D	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 270 cpm above background

(a) During operations with a potential for draining the reactor vessel.

~~(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in the secondary containment.~~

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment,~~
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Secondary containment inoperable ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	~~C.1 ---------NOTE--------- LCO 3.0.3 is not applicable. ------------------------------ Suspend movement of irradiated fuel assemblies in the secondary containment.~~	~~Immediately~~
	~~AND~~	
	~~C.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	C.~~3~~ 1 Initiate action to suspend OPDRVs.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

LCO 3.6.4.2 Each SCIV shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment,~~
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

-----NOTES-----

1. Penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIVs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one SCIV inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. <u>AND</u>	8 hours (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	~~D.1 ------NOTE------ LCO 3.0.3 is not applicable. -----------~~ ~~Suspend movement of irradiated fuel assemblies in the secondary containment.~~	~~Immediately~~
	~~AND~~	
	~~D.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	D.~~3~~ 1 Initiate action to suspend OPDRVs.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Three SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment,~~
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	~~NOTE~~ ~~LCO 3.0.3 is not applicable.~~		
	C.1	Place two OPERABLE SGT subsystems in operation.	Immediately
	<u>OR</u>		
	~~C.2.1~~	~~Suspend movement of irradiated fuel assemblies in secondary containment.~~	~~Immediately~~
		~~<u>AND</u>~~	
	~~C.2.2~~	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
		~~<u>AND</u>~~	
	C.2~~.3~~	Initiate action to suspend OPDRVs.	Immediately
D. Two or three SGT subsystems inoperable in MODE 1, 2, or 3.	D.1	Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two or three SGT subsystems inoperable ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	~~E.1 ----------NOTE---------- LCO 3.0.3 is not applicable. ------------------------------~~ ~~Suspend movement of irradiated fuel assemblies in secondary containment.~~	~~Immediately~~
	~~AND~~	
	~~E.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	E.~~3~~ 1 Initiate action to suspend OPDRVs.	Immediately

3.7 PLANT SYSTEMS

3.7.3 Control Room Emergency Ventilation (CREV) System

LCO 3.7.3 Two CREV subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment,~~
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CREV subsystem inoperable.	A.1	Restore CREV subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Be in MODE 4.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	C.1 ~~------NOTE------ LCO 3.0.3 is not applicable. -------------~~	
	Place OPERABLE CREV subsystem in pressurization mode.	Immediately
	<u>OR</u>	
	~~C.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.~~	~~Immediately~~
	~~<u>AND</u>~~	
	~~C.2.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~<u>AND</u>~~	
	C.2~~.3~~ Initiate action to suspend OPDRVs.	Immediately
D. Two CREV subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two CREV subsystems inoperable ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	~~E.1 ------------NOTE------------ LCO 3.0.3 is not applicable. --~~ ~~Suspend movement of irradiated fuel assemblies in the secondary containment.~~	~~Immediately~~
	~~AND~~	
	~~E.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	E.~~3~~ 1 Initiate action to suspend OPDRVs.	Immediately

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

~~b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ 1.0% when tested in accordance with ANSI N510-1975 at the system flowrate specified below, ± 10%.~~

~~ESF Ventilation System~~	~~Flowrate (cfm)~~
~~SGT System~~	~~9000~~
~~CREV System~~	~~3000~~

~~This testing shall be performed 1) every 24 months, 2) after partial or complete replacement of the charcoal adsorber bank, 3) after any structural maintenance on the system housing, or 4) following significant painting, fire, or chemical release in any ventilation zone communicating with the system.~~

~~c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, shows a methyl iodide efficiency ≥ 90% when tested in accordance with ASTM D3803-1989.~~

~~This testing shall be performed 1) every 24 months, 2) after every 720 hours of system operation, or 3) following significant painting, fire, or chemical release in any ventilation zone communicating with the system.~~

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

~~d~~ b. Once every 24 months demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below at the system flowrate specified below, ± 10%:

ESF Ventilation System	Delta P with charcoal adsorbers installed (inches water)	Delta P without charcoal adsorbers installed (inches water)	Flowrate (cfm)
SGT System	7	5	9000
CREV System	6	4	3000

~~e~~ c. Once every 24 months demonstrate that the heaters for the SGT System dissipate ≥ 40 kW when tested in accordance with ANSI N510-1975.

5.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained downstream of the offgas recombiners, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

a. The limits for concentrations of hydrogen downstream of the offgas recombiners and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

(continued)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, ~~and~~ 2, and 3.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify available volume of sodium pentaborate solution (SPB) is ≥ ~~3007~~ 4000 gallons.	24 hours
SR 3.1.7.2	Verify continuity of explosive charge.	31 days
SR 3.1.7.3	Verify the SPB concentration is ≥ 8.0% by weight	31 days AND Once within 24 hours after water or boron is added to solution
SR 3.1.7.~~3~~ 4	Verify the SPB concentration is ≤ 9.2% by weight.	31 days AND Once within 24 hours after water or boron is added to solution
	OR	
	Verify the concentration and temperature of boron in solution are within the limits of Figure 3.1.7-1.	Once within 8 hours after discovery that SPB concentration is > 9.2% by weight AND 12 hours

		thereafter
SR 3.1.7~~.4~~.5	Verify the minimum quantity of Boron-10 in the SLC solution tank and available for injection is ≥ 186 pounds.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.~~5~~6	Verify the SLC conditions satisfy the following equation: $\frac{(\ C\ )(\ Q\ )(\ E\ )}{(13\text{ wt. }\%)(86\text{ gpm})(19.8\text{ atom}\%)} \geq 1$ where, C = sodium pentaborate solution concentration (weight percent) Q = pump flow rate (gpm) E = Boron-10 enrichment (atom percent Boron-10)	31 days <u>AND</u> Once within 24 hours after water or boron is added to the solution
SR 3.1.7.~~6~~7	Verify each pump develops a flow rate ≥ 39 gpm at a discharge pressure ≥ 1325 psig.	24 months
SR 3.1.7.~~7~~8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 months on a STAGGERED TEST BASIS
SR 3.1.7.~~8~~9	Verify all piping between storage tank and pump suction is unblocked.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.~~9~~10	Verify sodium pentaborate enrichment is within the limits established by SR 3.1.7.~~5~~6 by calculating within 24 hours and verifying by analysis within 30 days.	24 months AND After addition to SLC tank
SR 3.1.7.~~10~~11	Verify each SLC subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position, or can be aligned to the correct position.	31 days

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low, Level 3	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≥ 528 inches above vessel zero
2. Drywell Pressure - High	1,2,3	2	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 2.5 psig
3. Reactor Zone Exhaust Radiation - High	1,2,3, (a)~~(b)~~	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 100 mR/hr
4. Refueling Floor Exhaust Radiation - High	1,2,3, (a)~~(b)~~	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 100 mR/hr

(a) During operations with a potential for draining the reactor vessel.

~~(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in secondary containment.~~

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Ventilation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low, Level 3	1,2,3,(a)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≥ 528 inches above vessel zero
2. Drywell Pressure - High	1,2,3	2	B	SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 2.5 psig
3. Reactor Zone Exhaust Radiation - High	1,2,3 (a)~~,(b)~~	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 100 mR/hr
4. Refueling Floor Exhaust Radiation - High	1,2,3, (a)~~,(b)~~	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 100 mR/hr
5. Control Room Air Supply Duct Radiation - High	1,2,3, (a)~~,(b)~~	1	D	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 270 cpm above background

(a) During operations with a potential for draining the reactor vessel.

~~(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in the secondary containment.~~

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment,~~
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Secondary containment inoperable ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	~~C.1 ------------NOTE------------ LCO 3.0.3 is not applicable. ------------------------------------ Suspend movement of irradiated fuel assemblies in the secondary containment.~~	~~Immediately~~
	~~AND~~	
	~~C.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	C.~~3~~ 1 Initiate action to suspend OPDRVs.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

LCO 3.6.4.2 Each SCIV shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment,~~
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

-----NOTES-----

1. Penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIVs.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more penetration flow paths with one SCIV inoperable.	A.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	8 hours
	<u>AND</u>		
			(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	~~D.1 ----------NOTE---------- LCO 3.0.3 is not applicable. ----------------------------~~ ~~Suspend movement of irradiated fuel assemblies in the secondary containment.~~	~~Immediately~~
	~~AND~~	
	~~D.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	D.~~3~~ 1 Initiate action to suspend OPDRVs.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Three SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment,~~
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	~~NOTE~~ ~~LCO 3.0.3 is not applicable.~~		
	C.1	Place two OPERABLE SGT subsystems in operation.	Immediately
	OR		
	~~C.2.1~~	~~Suspend movement of irradiated fuel assemblies in secondary containment.~~	~~Immediately~~
		~~AND~~	
	~~C.2.2~~	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
		~~AND~~	
	C.2~~.3~~	Initiate action to suspend OPDRVs.	Immediately
D. Two or three SGT subsystems inoperable in MODE 1, 2, or 3.	D.1	Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two or three SGT subsystems inoperable ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	~~E.1 ——NOTE——~~ ~~LCO 3.0.3 is not applicable.~~ ~~————————~~ ~~Suspend movement of irradiated fuel assemblies in secondary containment.~~	~~Immediately~~
	~~AND~~	
	~~E.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	E.~~3~~ 1 Initiate action to suspend OPDRVs.	Immediately

3.7 PLANT SYSTEMS

3.7.3 Control Room Emergency Ventilation (CREV) System

LCO 3.7.3 Two CREV subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment,~~
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CREV subsystem inoperable.	A.1	Restore CREV subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	C.1	~~-----NOTE-----~~ ~~LCO 3.0.3 is not applicable.~~ ~~-----------------~~ Place OPERABLE CREV subsystem in pressurization mode.	Immediately
	<u>OR</u>		
	~~C.2.1~~	~~Suspend movement of irradiated fuel assemblies in the secondary containment.~~	~~Immediately~~
	~~AND~~		
	~~C.2.2~~	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~		
	C.2~~.3~~	Initiate action to suspend OPDRVs.	Immediately
D. Two CREV subsystems inoperable in MODE 1, 2, or 3.	D.1	Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two CREV subsystems inoperable ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	~~E.1 ------NOTE------ LCO 3.0.3 is not applicable. ---------------------- Suspend movement of irradiated fuel assemblies in the secondary containment.~~	~~Immediately~~
	~~AND~~	
	~~E.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	E.~~3~~ 1 Initiate action to suspend OPDRVs.	Immediately

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

~~b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ 1.0% when tested in accordance with ANSI N510-1975 at the system flowrate specified below, ± 10%.~~

~~ESF Ventilation System~~	~~Flowrate (cfm)~~
~~SGT System~~	~~9000~~
~~CREV System~~	~~3000~~

~~This testing shall be performed 1) every 24 months, 2) after partial or complete replacement of the charcoal adsorber bank, 3) after any structural maintenance on the system housing, or 4) following significant painting, fire, or chemical release in any ventilation zone communicating with the system.~~

~~c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, shows a methyl iodide efficiency ≥ 90% when tested in accordance with ASTM D3803-1989.~~

~~This testing shall be performed 1) every 24 months, 2) after every 720 hours of system operation, or 3) following significant painting, fire, or chemical release in any ventilation zone communicating with the system.~~

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

~~d~~ b. Once every 24 months demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below at the system flowrate specified below, ± 10%:

ESF Ventilation System	Delta P with charcoal adsorbers installed (inches water)	Delta P without charcoal adsorbers installed (inches water)	Flowrate (cfm)
SGT System	7	5	9000
CREV System	6	4	3000

~~e~~ c. Once every 24 months demonstrate that the heaters for the SGT System dissipate ≥ 40 kW when tested in accordance with ANSI N510-1975.

5.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained downstream of the offgas recombiners, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

a. The limits for concentrations of hydrogen downstream of the offgas recombiners and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

(continued)

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

PROPOSED LICENSE AMENDMENT
ALTERNATIVE SOURCE TERM
MARKED PAGES - UFSAR

AFFECTED PAGE LIST

The following pages have been revised. On the affected pages the revised portions have been highlighted. A line has been drawn through the deleted text and a double underline for new or revised text.

UFSAR Pages

3.8-1 through 3.8-10

See Attached

3.8 STANDBY LIQUID CONTROL SYSTEM

3.8.1 Safety Objective

The safety objective of the Standby Liquid Control System is to provide a backup method, which is independent of the control rods, to make the reactor subcritical over its full range of operating conditions, and to provide sufficient buffering agent to maintain the suppression pool pH at or above 7.0 following a DBA LOCA involving fuel damage (see Section 14.6.3.5). Making the reactor subcritical is essential to permit the nuclear system to cool to the point where corrective actions can be carried out. ~~The system is designed for the normal modes of cooldown only; it is not intended to function after a loss-of-coolant accident.~~ Maintaining the suppression pool pH at or above 7.0 following a LOCA involving fuel damage supports the LOCA radiological dose analyses that do not consider the re-evolution of iodine to the containment atmosphere.

3.8.2 Safety Design Basis

1. Backup capability for reactivity control shall be provided, independent of normal reactivity control provisions in the nuclear reactor, to shut down the reactor if the normal control is impaired so that cold shutdown (MODE 4) cannot be obtained with control rods alone.

2. The backup system shall have the capacity for controlling the reactivity difference between the steady-state rated operating condition of the reactor and the cold shutdown condition (MODE 4), including shutdown margin, to assure complete shutdown from the most reactive condition at any time in the core life.

3. The time required for actuation and effectiveness of the backup reactivity control shall be consistent with the nuclear reactivity rate of change predicted between rated operating and cold shutdown conditions (MODE 4). A scram of the reactor or operational control of fast reactivity transients is not specified to be accomplished by this system.

4. Means shall be provided by which the functional performance capability of the ~~backup control~~ system components can be verified periodically under conditions approaching actual use requirements. Demineralized water, rather than the actual neutron absorber solution, is injected into the reactor to test the operation of all components of the redundant control system.

5. The neutron absorber shall be dispersed within the reactor core in sufficient quantity to provide a reasonable margin for leakage, dilution, or imperfect mixing.

6. The system shall be reliable to a degree consistent with its role as a special safety system.

7. The possibility of unintentional or accidental shutdown of the reactor by this system shall be minimized.

8. The system shall be capable of supplying buffering agent to the suppression pool in the event of a large recirculation break. Sufficient buffering agent shall be provided to ensure that the pH of the suppression pool for DBA post-LOCA events involving fuel damage remains at or above 7.0 for 30 days.

3.8.3 Description (Figures 3.8-1, 3.8-2, 3.8-3, 3.8-5, and 3.8-6)

The Standby Liquid Control System is manually initiated from the Main Control Room to pump a boron neutron absorber solution into the reactor if:

1. the operator determines the reactor cannot be shut down or kept shut down with the control rods, or

2. fuel damage occurs post-LOCA.

The Standby Liquid Control System is required ~~only~~ to shut down the reactor at a steady rate within the capacity of the shutdown cooling systems and to keep the reactor from going critical again as it cools.

The Standby Liquid Control System is needed ~~only~~ in the improbable event that not enough control rods can be inserted in the reactor core to accomplish subcriticality in the normal manner.

The Standby Liquid Control System is also required to supply sodium pentaborate solution for post-LOCA events that involve fuel damage to maintain the suppression pool pH at or above 7.0. The radiological dose analysis for the DBA LOCA assumes concentrations of iodine species consistent with a suppression pool pH at or above 7.0 (i.e., re-evolution of iodine to the containment atmosphere is not considered). The sodium pentaborate solution is credited as a buffering agent to offset the post-LOCA production of acids (e.g., radiolysis products).

The system consists of a boron solution tank, a test water tank, two positive-displacement pumps, two explosive-actuated valves, and associated local valves and controls. They are mounted in the Reactor Building outside the primary containment. The liquid is piped into the reactor vessel via the differential pressure and liquid control line and discharged near the bottom of the core lower support plate through a standpipe so it mixes with the cooling water rising through the core (see Sections 4.2, "Reactor Vessel and Appurtenances Mechanical Design," and 3.3, "Reactor Vessel Internals Mechanical Design").

The Boron-10 isotope absorbs thermal neutrons and thereby terminates the nuclear fission chain reaction in the uranium fuel.

The specified neutron absorber solution is enriched sodium pentaborate ($Na_2B_{10}0_{16}$-$10H_20$). It consists of a mixture of borax, enriched boric acid, and demineralized water prepared in accordance with approved plant procedures to ensure the proper volume and enriched sodium pentaborate concentration is present in the standby liquid control tank. A sparger is provided in the tank for mixing, using air. To prevent system plugging, the tank outlet is raised above the bottom of the tank and is fitted with a strainer.

At all times when it is possible to make the reactor critical, the configuration of the Standby Liquid Control System shall satisfy the following equation:

$$\frac{(\ C\ )(\ Q\ )(\ E\ )}{(\ 13\ \text{WT\%}\ )\ (\ 86\ \text{GPM}\ )\ (\ 19.8\ \text{ATOM\%}\ )} \geq 1.0$$

C = sodium pentaborate solution weight percent concentration
Q = SLCS pump flow rate in gpm
E = Boron-10 atom percent enrichment in the sodium pentaborate solution

The SLC system is used to control Suppression Pool pH in the event of a DBA LOCA by injecting sodium pentaborate into the reactor vessel. It is them transported to the suppression pool and mixed by ECCS flow circulating through the reactor, out of the recirculation break and into the suppression chamber. The amount of sodium pentaborate solution that must be available for injection following a DBA LOCA is determined as part of the DBA LOCA Radiological analysis. This quantity is maintained in the storage tank as specified in the Technical Specifications.

The solution concentration is normally limited to a maximum of 9.2 weight percent to preclude unwanted precipitation of the sodium pentaborate. The saturation temperature of the 9.2 percent solution is 40°F which provides a 10°F thermal margin below the lowest temperature predicted for the SLCS equipment area. Tank heating components provide backup assurance that the sodium pentaborate solution temperature will never fall below 50°. The sodium pentaborate solution concentration is allowed to be ☺9.2 weight percent provided the concentration and temperature of the solution are within the limits permitted by the technical specifications. High or low temperature, high or low liquid level, or a shorted heater causes an alarm in the control room. Tank level indication is also provided in the control room.

Each positive displacement pump is sized to inject the solution into the reactor in 50 to 125 minutes (approximately 50 gpm), depending on the amount of solution in the tank, at the reactor vessel maximum operating pressure. The pump and system design pressure is 1500 psig. The two relief valves are set at approximately 1425 psig to exceed the reactor operating pressure by a sufficient margin to avoid valve leakage.

To prevent bypass flow from one pump in case of relief valve failure in the line from the other pump, a check valve is installed downstream of each relief valve line in the pump discharge pipe.

A bladder-type pneumatic-hydraulic accumulator is installed on the piping near each relief valve to dampen pulsations from the pumps to protect the system.

The two explosive-actuated injection valves provide high assurance of opening when needed and ensure that the boron solution will not leak into the reactor even when the pumps are being tested. The valves have a demonstrated firing reliability in excess of 99.99 percent. Each explosive valve is closed by a plug in the inlet chamber. The plug is circumscribed with a deep groove so the end will readily shear off when pushed with the valve plunger. This opens the inlet hole through the plug. The sheared end is pushed out of the way in the chamber and is shaped so it will not block the ports after release.

The shearing plunger is actuated by an explosive charge with dual ignition primers inserted in the side chamber of the valve. Ignition circuit continuity is monitored by a trickle current, and an alarm occurs in the control room if either circuit opens. Indicator lights show which primer circuit is opened. To service a valve after firing, a 6-inch length of pipe (spool piece) must be removed immediately upstream of the valve to gain access to the shear plug.

The Standby Liquid Control System is actuated by a five-position spring return to "normal" keylock switch located on the control room console. The keylock feature ensures that switching from the "stop" position is a deliberate act (safety design basis 7). Momentarily placing the switch to either "start A" or "start B" position starts the respective injection pump, opens both explosive valves, and closes the Reactor Water Cleanup System isolation valves to prevent loss or dilution of the boron solution.

A green light in the control room indicates that power is available to the pump motor contactor, but that the contactor is open (pump not running). A red light indicates the contactor is closed (pump running). A white light indicates that the motor has tripped or the local handswitch is in the test position.

A red light beside the switch turns on when liquid is flowing through an orifice flow switch downstream of the explosive valves. If the flow light or pump lights indicate that the liquid may not be flowing, the operator can immediately turn the switch to the other side, which actuates the alternate pump. Crosspiping and check valves assure a flow path through either pump and either explosive valve. The chosen pump will start even though its local switch at the pump is in the "stop" position for test or maintenance. Pump discharge pressure indication is also provided in the control room.

Equipment drains and tank overflow are piped not to the waste system but to separate containers (such as 55-gallon drums) that can be removed and disposed of independently to prevent any trace of the boron solution from inadvertently reaching the reactor.

Instrumentation is provided locally at the standby liquid control tank consisting of solution temperature indication and control, tank level, and heater status. Instrumentation and control logic is presented in Figures 3.8-4 and 3.8-7, Mechanical Logic Diagram.

3.8.4 Safety Evaluation

3.8.4.1 Reactivity Control

The Standby Liquid Control System is a special safety system not required for normal plant operation, and is never expected to be needed for reactor shutdown because of the large number of control rods available to shut down the reactor.

~~To assure the availability of the Standby Liquid Control System, two sets of the components required to actuate—the pumps and explosive valves—are provided in parallel redundancy (safety design basis 6).~~

The system is designed to make the reactor subcritical from rated power to a cold shutdown (MODE 4) at any time in core life. The reactivity compensation provided will reduce reactor power from rated to the after-heat level and allow cooling the nuclear system to normal temperature with the control rods remaining withdrawn in the rated power pattern. It includes the reactivity gains due to complete decay of the rated power xenon inventory. It also includes the positive reactivity effects from eliminating steam voids, changing water density from hot to cold, reduced Doppler effect in uranium, reduction of neutron leakage from the boiling to cold condition, and decreasing control rod worth as the moderator cools. A licensing analysis is performed each cycle to verify adequate SLCS shutdown capacity. The analysis assumes the specified minimum final concentration of boron in the reactor core and allows for calculational uncertainties. The SLCS shutdown capacity is reported in Appendix N.

The specified minimum average concentration of natural boron in the reactor to provide the specified shutdown margin, after operation of the Standby Liquid Control System, is 660 ppm (parts per million). The minimum quantity of sodium pentaborate to be injected into the reactor is calculated based on the required 660 ppm average concentration in the reactor coolant, Boron-10 enrichment, the quantity of reactor coolant in the reactor vessel, recirculation loops, and the entire RHR System in the shutdown cooling mode, at 70°F and reactor normal water level. The result is increased by 25 percent to allow for imperfect mixing, leakage, and volume in other piping connected to the reactor. This minimum concentration is achieved by preparing the solution as defined in paragraph 3.8.3 and maintaining it above saturation temperature. This satisfies safety design basis 5.

Cooldown of the nuclear system will take several hours, at a minimum, to remove the thermal energy stored in the reactor, cooling water, and associated equipment, and to remove most of the radioactive decay heat. The controlled limit for the reactor coolant temperature cooldown is 100°F per hour. Normal operating temperature is about 550°F. Usually, shutting down the plant with the main condenser and various shutdown cooling systems will take 10 to 24 hours before the reactor vessel is opened, and much longer to reach room temperature (70°F). Room temperature is the condition of maximum reactivity and, therefore, the condition which requires the maximum boron concentration. Thus safety design basis 2 is met.

The specified boron injection rate is limited to the range of 7 to 40 ppm per minute change of boron concentration in the reactor pressure vessel and recirculation loop piping water volumes. The lower rate ensures that the boron is injected into the reactor in less than 2 hours, which is considerably faster than the cooldown rate. The upper limit injection rate insures that there is sufficient mixing such that the boron does not recirculate through the core in uneven concentrations which could possibly cause asymmetric power oscillations in the core. This satisfies safety design basis 3.

3.8.4.2 Suppression Pool pH Control

The Standby Liquid Control System is required to supply sodium pentaborate solution for post-LOCA events that involve fuel damage to maintain the suppression pool pH at or above 7.0. The radiological dose analysis for the DBA LOCA assumes concentrations of iodine species consistent with a suppression pool pH at or above 7.0 (i.e., re-evolution of iodine to the containment atmosphere is not considered).

The quantity of sodium pentaborate necessary to offset the post-LOCA production of acid and maintain the suppression pool pH at or above 7.0 has been documented as part of the LOCA radiological dose analysis. This quantity is maintained in the storage tank as specified in the technical specifications. Maintaining the suppression pool pH at or above 7.0 is a concern following a DBA LOCA involving fuel damage. With a LOCA involving a recirculation pipe break, there will be sufficient flow from the ECCS systems through the reactor vessel and out of the break to transport the buffering agent to the suppression pool. The calculation methodology for suppression pool pH control was based on the approach outlined in NUREG-1465 and NUREG/CR-5950. The pH of the suppression pool water was calculated using the STARpH code. The design inputs were conservatively established to maximize the post-LOCA production of acids and to minimize the post-LOCA production and/or addition of bases. Other design input values such as initial suppression pool volume and pH were selected to minimize the calculated pH.

It is expected that the initial effects on post-accident suppression pool pH will come from rapid fission product transport and the formation of cesium compounds, which would result in increasing the suppression pool pH. However, cesium compounds are not credited in the long-term pH analyses and the determination of the final (30 day) pH value. As radiolytic production of nitric acid and hydrochloric acid proceeds, and these acids are transported to the pool over the first days of the event, the pH would become more acidic.

The buffering effect of SLC injection within several hours is sufficient to offset the affects of these acids that are transported to the suppression pool. In these events, the addition of a buffering agent to the suppression pool offsets the radiolysis production of acids. This satisfies safety design basis 8.

3.8.4.3 Common Safety Evaluation

The Standby Liquid Control System is classified as a special safety system. To assure the availability of the Standby Liquid Control System, two sets of the components required to actuate --the pumps and explosive valves-- are provided in parallel for redundancy (safety design basis 6). The system is also credited for limiting radiological dose following a DBA LOCA as described in Section 14.6.3.5, in accordance with the AST analyses for suppression pool pH control; however, the system will remain classified as a special safety system instead of being classified as a safety system.

The Standby Liquid Control System is designed as a Class I system for withstanding the specified earthquake loadings (see Appendix C). Nonprocess equipment such as the test tank is designed as Class II. The system piping and equipment are designed, installed, and tested in accordance with USAS B31.1.0, Section I.

The Standby Liquid Control System is not required to be designed to meet the single failure criterion because it serves as a backup to the control rods. System reliability is enhanced by providing redundancy of pumps and valves. Hence, redundancy is not required for the tank heater or the heating cable.

The Standby Liquid Control System is required to be operable in the event of a station power failure so the pumps, valves, and controls are powered from the standby AC power supply in the absence of normal power. The pumps and valves are powered and controlled from separate buses and circuits so that a single failure will not prevent system operation. The essential instruments and lights are powered from the 120-V AC instrument power supply.

The Standby Liquid Control System and pumps have sufficient pressure margin, up to the system relief valve nominal setting of 1425 psig, to assure solution injection into the reactor at a pressure of at least three percent above the lowest setpoint of the main steam relief valves (1140 psig pre-uprated; 1174 psig uprated). The nuclear system is protected from overpressurization during operation of the Standby Liquid Control System positive displacements pumps by the nuclear system main steam relief valves.

Only one of the two standby liquid control pumps is needed for proper system operation. If one pump is inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. The system pumps are powered by a diesel backed source and are not load shed. The period during which one redundant component upstream of the explosive valves may be out of operation will be consistent with the very small probability of failure of both the control rod shutdown capability and the alternate component in the Standby Liquid Control System, together with the fact that nuclear system cooldown takes 10 or more hours while liquid control solution injection takes about 2 hours. For the Standby Liquid Control System suppression pool pH function, this condition will be consistent with other two train systems. Although the Standby Liquid Control System is not strictly a two train system, the redundancies in active components most susceptible to failure (e.g., pumps and squib valves) provide additional assurance that most single failures will not impede the ability to perform its function. This indicates the considerable time available for testing and restoring the Standby Liquid Control System to operable condition after testing while reactor operation continues. Assurance that the system will still fulfill its function during repairs is obtained by demonstrating operation of the operable pump.

It can be seen that the Standby Liquid Control System satisfies safety design basis 1.

3.8.5 Inspection and Testing

Operational testing of the Standby Liquid Control System is performed in at least two parts to avoid injecting boron into the reactor inadvertently. By opening two closed valves to the solution tank, the boron solution may be recirculated by turning on either pump with its local switch. With the valves to and from the solution tank closed and the three valves opened to and from the test tank, the demineralized water in the test tank can be recirculated by turning on either pump locally. After pumping boron solution, demineralized water is pumped to flush out the pumps and pipes. Functional testing of the injection portion of the system is accomplished by closing the open valve from the solution tank, opening the closed valve from the test tank, and actuating the switch in the control room to either the A or B circuit. This starts one pump and ignites one of the explosive actuated injection valves to open. The lights and alarms in the control room indicate that the system is functioning. This satisfies safety design basis 4.

After the functional test, the affected injection valve and explosive charge must be replaced and all the valves returned to their normal positions as indicated in Figures 3.8-1, 3.8-2, 3.8-3, 3.8-5, and 3.8-6.

By closing a local normally open valve to the reactor in the containment, leakage through the injection valves can be detected at a test connection in the line between the containment isolation check valves. (A position indicator light in the control room indicates when the local valve is full open and ready for operation.) Leakage from the reactor through the first check valve can be detected by opening the same test connection whenever the reactor is pressurized.

The test tank contains sufficient demineralized water for testing pump operation. Demineralized water from the makeup or condensate storage system is available at 30 gpm for refilling or flushing the system.

Should the boron solution ever be injected into the reactor, either intentionally or inadvertently, then after making certain that the normal reactivity controls will keep the reactor subcritical, the boron is removed from the reactor coolant system by flushing for gross dilution followed by operation of the reactor cleanup system. There is practically no effect on reactor operations when the boron concentration has been reduced below about 50 ppm.

The sodium pentaborate solution weight percent in the SLCS storage tank is periodically determined by titration or equivalent chemical analysis. The Boron-10 isotopic atom percent concentration of the solution is also determined periodically, utilizing mass spectrometry or equivalent technology.

The gas pressure in the two accumulators is measured periodically to detect leakage. A pressure gauge and portable nitrogen supply are required to test and recharge the accumulators.

ENCLOSURE 4

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

PROPOSED LICENSE AMENDMENT
ALTERNATIVE SOURCE TERM
SAFETY ASSESSMENT

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

ACRONYMS AND ABBREVIATIONS

µCi/gm	micro-curies per gram
" HG	inches of mercury
AST	Alternative Source Term
BFN	Browns Ferry Nuclear Plant
BLEU	Blended Low Enriched Uranium
BWR	Boiling Water Reactor
CAD	Containment Air Dilution
cfm	cubic feet per minute
CREV	Control Room Emergency Ventilation
CsI	Cesium Iodine
DBA	Design Basis Accident
DE	Dose Equivalent
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EPU	Extended Power Uprate
ft	feet
GDC	General Design Criterion
GE	General Electric
gpm	gallons per minute
H+	Hydrogen Ion
HEPA	High Efficiency Particulate Air
HNO_3	Nitric Acid
hrs	hours
HWWV	Hardened Wetwell Vent
in	inch
lbm	pounds-mass
LOCA	Loss of Coolant Accident
LPZ	Low Population Zone
m/s	meters per second

ACRONYMS AND ABBREVIATIONS

MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
MWt	Megawatt thermal
OH-	Hydroxyl Ions
PCIS	Primary Containment Isolation Signal
pH	Hydrogen Ion Concentration
psid	pounds per square inch differential
rem	roentgen equivalent man
RG	Regulatory Guide
scfh	standard cubic feet per hour
SGT	Standby Gas Treatment System
secs	seconds
SER	Safety Evaluation Report
SLC	Standby Liquid Control System
SPB	Sodium Pentaborate
SR	Surveillance Requirement
SRP	Standard Review Plan
TEDE	Total Effective Dose Equivalent
TS	Technical Specification
TVA	Tennessee Valley Authority
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
X/Q	Atmospheric Dispersion Factor

1. INTRODUCTION

1.1 Evaluation Overview and Objective

The objective of this safety assessment is to document BFN implementation of the Alternative Source Term (AST). The implementation of AST is governed by 10 CFR 50.67, the guidelines of the Standard Review Plan (SRP) Section 15.0.1 (Reference 1), and Regulatory Guide (RG) 1.183 (Reference 2).

BFN has elected to perform a full scope implementation of the AST as defined in RG 1.183. The implementation consists of the following:

1. Identification of the core source term based on plant specific analysis of core fission product inventory.
2. Determination of the release fractions for the four Updated Final Safety Analysis Report (UFSAR) Boiling Water Reactor (BWR) Design Basis Accidents (DBAs) that could potentially result in control room and offsite doses. These are the loss of coolant accident (LOCA), the main steam line break accident, the refueling accident, and the control rod drop accident.
3. Calculation of fission product deposition rates and removal efficiencies.
4. Calculation of offsite and control room personnel Total Effective Dose Equivalent (TEDE).
5. Evaluation of suppression pool pH to ensure that the particulate iodine deposited into the suppression pool during a DBA LOCA does not re-evolve and become airborne as elemental iodine.
6. Calculation of a new control room atmospheric dispersion factor (X/Q) for a main steam line break accident instantaneous ground level puff release.
7. Evaluation of other related design and licensing bases such as NUREG-0737 (Reference 3).

The radiological dose analyses have been performed assuming reactor operation at a thermal power of 4031 MWt (102% of 3952 MWt). This results in a conservative estimate of fission product releases for operation at current licensed power of 3458 MWt.

1.2 Major Aspects of AST Analyses

Implementation of AST includes changes to the methodology presently used at BFN. These include:

1. Development of a bounding plant-specific core fission product inventory.

2. Analysis of a new X/Q for an instantaneous ground level puff release to the atmosphere for the main steam line break accident.

3. No credit is taken for Control Room Emergency Ventilation (CREV) System or standby gas treatment (SGT) System charcoal adsorption for any DBA.

4. No credit is taken for CREV System or SGT System HEPA filter particulate removal for any DBA except LOCA.

5. New requirements were developed for post-LOCA standby liquid control (SLC) System operation for suppression pool pH control along with calculation of minimum sodium penteborate quantity requirements.

1.3 Summary

Implementation of the AST as the plant radiological consequence analyses licensing basis requires a license amendment per the requirements of 10 CFR 50.67. The enclosed AST analyses demonstrate the offsite and control room post-accident radiological doses remain within regulatory limits.

2. EVALUATION

2.1 Scope

2.1.1 Accident Radiological Consequence Analyses

The DBA accident analyses documented in Chapter 14 of the BFN UFSAR (Reference 4) that could potentially result in control room and offsite doses were addressed using methods and input assumptions consistent with the AST. The following DBAs were addressed:

- LOCA, UFSAR Section 14.6.3
- Main Steam Line Break Accident, UFSAR Section 14.6.5
- Refueling Accident, UFSAR Section 14.6.4
- Control Rod Drop Accident, UFSAR Section 14.6.2

The analysis was performed per RG 1.183. The results were evaluated to confirm compliance with the acceptance criteria presented in 10 CFR 50.67 and GDC 19 of 10 CFR 50, Appendix A. Computer codes used in the DBA analyses results are listed in Table 2-1.

The AST control room dose analyses are applicable for all three unit control rooms. The Unit 1 and 2 control rooms are shared in a common room with Unit 1 at one end and Unit 2 at the other. The Unit 3 control room, though separated from the Units 1 and 2 control room, is part of the same control bay habitability zone.

The refuel zone is a common three-unit zone consequently; the refueling accident radiological consequence analysis is the only analyses applicable to all three units. Since the Unit 1 is in an extended shutdown, the remaining three DBA radiological consequence analyses have not been performed for Unit 1. However, TVA expects that the results will be similar to Units 2 and 3.

2.1.2 Suppression Pool pH Control

A calculation was performed to evaluate the suppression pool pH in the event of a DBA LOCA. The objective of the analysis was to demonstrate that the suppression pool pH remains at or above 7.0, thus ensuring that the particulate iodine (cesium iodide - CsI) deposited into the suppression pool during this event does not re-evolve and become airborne as elemental iodine. The analysis credits the pH buffering effect of sodium pentaborate introduced into the suppression pool post-LOCA by SLC operation to maintain the pH above 7.0.

2.1.3 Main Steam Line Break Accident Puff Release Dispersion Factor

A new control room X/Q was determined for use in the main steam line break accident analysis. This X/Q reflects an instantaneous ground level puff release to the atmosphere in accordance with Regulatory Guide 1.183 Appendix D.

2.1.4 NUREG-0737 Evaluation

An evaluation was performed to identify potential impacts of applying AST methodologies on compliance with NUREG-0737 requirements. This evaluation included the following:

- Revision of the current radiological dose analyses for post-accident vital area access and post-accident sampling (NUREG-0737, Item II.B.2 and Item II.B.3),
- Revision of the current radiological dose analyses for the post-accident containment high range radiation monitors (NUREG-0737, Item II.F.1),
- Revision of control room post-accident radiological dose analyses for emergency support facility upgrades and control room habitability (NUREG-0737, Items III.A.1.2 and III.D.3.4), and
- Consideration of post-accident sources of radiation and radioactivity outside the primary containment in terms of impact on dose analysis related to integrity of systems outside containment likely to contain radioactive material (NUREG-0737, Item III.D.1.1).

2.1.5 Environmental Qualification

The radiation doses used for the environmental qualification analyses at the original licensed thermal power conditions were calculated using source terms determined by TID-14844 (Reference 5) methodology. The radiation doses used for the environmental qualification analyses at both current licensed thermal power and Extended Power Uprate (EPU) conditions are adjusted upward from the original values based on the determined source terms of the ORIGEN computer code for the respective power level.

2.2 Method of Evaluation

2.2.1 Accident Radiological Consequence Analyses

Analyses were prepared for the simulation of the radionuclide release, transport, removal, and doses estimated for the postulated accidents listed in Section 2.1.1.

The ORIGEN code (Reference 6) was used to calculate plant-specific fission product inventories which bound the effect of two-year fuel cycles, power operation at EPU conditions (4031 MWt (102% of 3952 MWt)), and using current and anticipated fuel designs. The fission product inventory for General Electric (GE)-14, Framatome Atrium-10 fuel, and Framatome Blended Low Enriched Uranium (BLEU) fuel designs were evaluated. Bounding values of fission product activity were determined for each radionuclide in the DBA radiological analyses. Fission product activities were calculated for immediately after shutdown and 24 hours following shutdown. The values are shown in Table 2-2.

The RADTRAD computer code Version 3.02(a) (Reference 7) was used for the DBA dose calculations. The computer code STARDOSE (Reference 8) was used to check the RADTRAD results. The RADTRAD and STARDOSE programs are radiological consequence analysis codes used to determine post-accident doses at offsite and control room locations. The STARDOSE code is the proprietary property of Polestar Applied Technology, Inc., and the NRC has previously reviewed results obtained from the application of this code.

The existing UFSAR X/Q values were developed prior to and used in support of the license amendment request (References 9 and 10) for increased main steam isolation valve (MSIV) leakage rate limits. Control room X/Q values for the base of the stack releases were calculated using the computer code ARCON96 (Reference 11). For sites such as BFN with control room ventilation intakes that are close to the base of tall stacks, ARCON96 underpredicts the X/Q values for top of stack releases; therefore, top of stack releases to the control room intakes were evaluated using the methods of Regulatory Guides 1.111 (Reference 12) and 1.145 (Reference 13). The X/Q values associated with top of stack, base of stack, and turbine building releases were reviewed by the NRC in the Safety Evaluation for Amendments 263 and 223 for BFN Units 2 and 3, respectively (Reference 14). The existing X/Q values applicable to the time periods, distances, and geometric relationships are shown in Tables 2-3 through 2-7. Existing values for X/Q were used for AST radiological dose analyses except for the establishment of a new control room X/Q value associated with an instantaneous ground level puff release for the case of a main steam line break accident (see Section 2.2.3).

The post-LOCA shine dose to personnel in the control room includes the radiation shine from the secondary containment airborne activity and gamma dose from Core Spray System piping, which is in close proximity to the control building. An evaluation was performed of the existing TID-14844 analysis to determine application values for AST. For radiation from the Core Spray System piping, a comparison of gamma radiation plots from the suppression pool water was performed for high energy photons to determine similarity of shapes for the TID-14844 source term and the AST source term.

For the secondary containment airborne shine dose, a shine dose multiplier for AST airborne radioiodines was developed to enable direct comparison of the TID-14844 and the AST shine dose. To support this comparison, the activity for TID-14844 was increased to account for the increase in power level. The resulting comparison of several key nuclides found that the AST I-131 and I-133 activities in the reactor building are approximately a factor of 3 lower at 1.3 hours and 5 hours and, a factor of 30 lower at 24 hours compared to the TID-14844 levels at the same times. Considering the highest multiplier for the AST radionuclides (used to account for the activity other than iodine, especially for cesium) for 1 to 8 hours and at 24 hours, the effective iodine activity airborne in the reactor building for AST would be about the same before 8 hours and about a factor of 10 lower at 24 hours compared to TID-14844. For noble gases, the AST activities are about a factor of two lower than the TID-14844 source term at two hours, and by 24 hours, they are about the same.

The evaluation established that the integrated gamma dose from Core Spray System piping is slightly higher than previous over the 720 hours duration of the accident for the AST. However, only about 25 percent of the total 720 hour control room shine dose is due to the Core Spray System piping contribution. The control room shine dose from airborne activity in the secondary containment will be substantially reduced for the AST as compared to the TID-14844 source term. Therefore, the existing integrated control room shine dose, even if increased by the power ratio of EPU, is acceptable for a combination of EPU operation and AST application. This evaluation was checked using the MicroShield code, Version 5.03 (Reference 15). MicroShield is a point kernel integration code used for general purpose gamma shielding analysis. MicroShield has been used in safety-related applications by many nuclear plants in the United States. In this BFN application, it has been used as a means for design verification as an independent analysis.

For the main steam line break accident, radiation shine from the turbine building was conservatively handled assuming all released inventory is in the turbine building for two hours. Radiation shine from the airborne activity having escaped the turbine building is handled explicitly by the TVA computer code COROD. The calculation incorporates the control building dimensions and concrete roof (2.25 ft thick) in conjunction with the main steam line break accident released radioisotopes in a cloud above the control building.

2.2.2 Suppression Pool pH Control Calculations

The calculation methodology for suppression pool pH control was based on the approach outlined in NUREG-1465 (Reference 16) and NUREG/CR-5950, (Reference 17). Specifically, credit was taken for sodium pentaborate (SPB) addition to the suppression pool water as a result of SLC operation. The pH of the suppression pool water was then calculated using the STARpH code (Reference 18). This same methodology and code for calculation of transient suppression pool pH (including the formation of acids by radiation effects on drywell components) was applied to the Hope Creek AST application (Reference 19).

Calculations were performed to verify sufficient SPB solution is available to maintain the suppression pool pH at or above 7.0 for 30 days post accident. The design inputs were conservatively established to maximize the post-LOCA production of acids and to minimize the post-LOCA production and/or addition of bases. Other design input values such as initial suppression pool volume and pH were selected to minimize the calculated pH. It was determined that the calculated required quantity of SPB was in excess of the current TS limit. Therefore, a change to TS 3.1.7, Standby Liquid Control System (SLC), is being proposed increase the required amount of SPB.

2.2.3 Main Steam Line Break Accident Instantaneous Ground Level Puff Release Dispersion Factor

To meet RG 1.183 requirements for a main steam line break a new X/Q for a puff release was calculated. An instantaneous ground level puff release is assumed. The calculation of the main steam line break accident ground level puff release dispersion factor uses plant parameters for the main steam line break accident (e.g., mass of liquid-steam mixture released, timing of release, temperature of the liquid-steam mixture) to obtain the initial conditions of the released steam puff. The steam puff is treated as a "bubble" with a given transit time up to and across the control room intake. Once introduced into the atmosphere, the steam bubble rises at a rate corresponding to the buoyancy force (resulting from the density difference between ambient air and hot steam) equaling the drag force resulting from the friction between the bubble mass and the surrounding air. Mixing of the steam with surrounding air reduces the bubble's buoyancy, but also increases dilution. Different bubble shapes and degrees of air entrainment are considered, and the worst case is used (i.e., minimum dilution). No credit is taken for concentration gradients within the rising bubble. In particular, no credit is taken for a vertical concentration gradient; (i.e., the concentration at the elevation of the control room air intake is assumed to be the same as that of the leading edge of the rising bubble).

The bubble is assumed to be released from the turbine building at a distance from the nearest control room intake that is exceeded by 90% of the potential release locations. No credit is taken for wind direction; (i.e., it is assumed that the centerline of the bubble trajectory always passes over one control room intake). The diameter of the bubble (even with substantial initial air entrainment) is less than the distance between the two air intakes

The minimum dilution effect was quantified as a dilution factor of 0.25; (i.e., a factor of four decrease in the initial concentration of activity in the release). This is an average value during the time of passage over the control room air intake.

2.2.4 NUREG-0737 Evaluation

- Post Accident Vital Area Access and Sampling - Post-accident personnel missions resulting in mission doses (including post-accident sampling) were identified. Plant calculations used in support of plant post-accident vital area access (prepared in accordance with the requirements of NUREG-0737, Item II.B.2 and II.B.3) were revised for impact by AST. The revisions considered the comparative radiation levels from AST and the existing TID-14844 methodology source terms (such as airborne activity in the reactor building and turbine building, and also as activity in the suppression pool water).
- Post-Accident Radiation Monitor - Post-accident containment high range radiation monitoring calculations were revised for impact by AST (NUREG-0737, Item II.F.1).
- Control Room Radiation Protection - The control room radiological dose impact of AST has been specifically calculated for each of the four DBAs analyzed for AST implementation (NUREG-0737, Item III.D.3.4).
- Radioactive Sources Outside the Primary Containment - The DBA LOCA control room dose analysis, as well as that for offsite doses, includes the effects of coolant leakage outside the primary containment and (for the control room dose analyses only) the shine contribution from Core Spray System piping (NUREG - 0737, Item III.D.1.1).

2.3 Inputs and Assumptions

2.3.1 Accident Radiological Consequence Analyses

For AST accident radiological consequences, analyses were performed for the four DBAs that could potentially result in control room and offsite doses. These are the LOCA, main steam line break accident, refueling accident, and control rod drop accident.

Plant-specific fuel design parameters were used in the fission product and transuranic nuclide inventories for the accident analyses. Table 2-8 summarizes key fuel cycle parameters.

The reactor core inventory of activity for the AST dose analyses is based on an average burn-up of 35 to 37 GWd/MT depending on the fuel type. For the control rod drop accident and refueling accident analyses, a RG 1.183 minimum core radial peaking factor of 1.5 was used. For the refueling accident analysis, the core isotopic inventory after 24 hours of decay was used.

The release source term is developed using the radionuclide isotopes listed in Table 2-2 and the release fractions from Table 1 of RG 1.183. The radionuclides that are included are those identified as being potentially important contributors to TEDE in NUREG/CR-6604 (Reference 7). Release fractions for LOCA as release rates are shown in Table 2-9.

No credit is taken for the adsorption of elemental iodine, organic iodine, or noble gases by charcoal in the SGT or CREV systems for any of the four DBAs. SGT and CREV system HEPA filters are credited for removal of 90 percent of the particulate activity in the LOCA analysis. HEPA filter removal of particulates activity was not credited for the remaining three DBA analyses. A comparison of CREV/SGT functions modeled in the AST radiological dose analyses is presented in Table 2-10.

The CREV System is automatically initiated by a Group 6 primary containment isolation signal (PCIS), by high radiation at the control bay air intakes, or it can be manually initiated by the control room operators. The PCIS Group 6 trip signal is initiated by reactor vessel low water level, drywell high pressure, or reactor building ventilation high radiation. For a LOCA, the PCIS Group 6 initiation of CREV will occur significantly prior to the control room experiencing conditions which would result in excessive doses to the control room operators and, hence, significantly prior to an initiation on control bay air intake high radiation. The adequacy of the control bay radiation monitoring setpoint was reviewed as part of the AST NUREG-0737 evaluation.

In accordance with Standard Review Plan Section 6.4 (Reference 25), the doses due to airborne activity released from the turbine building may be divided by a factor of two because the CREV intakes are on opposite sides of the building and the makeup flow is equal from each intake.

The BFN Emergency Core Cooling Systems (ECCS) are designed, maintained, and tested to minimize the radiological consequences following a postulated DBA. The AST analyses inputs and assumptions are consistent with the design and licensing for these systems.

An assumed unfiltered inleakage rate of 3717 cubic feet per minute into the control room habitability zone was used. This inleakage rate was acknowledged by NRC in the Safety Evaluation for Amendments 263 and 223 for BFN Units 2 and 3, respectively (Reference 14).

The standard breathing rates specified in RG 1.183 have been used. The key accident radiological consequence analyses inputs are summarized in Table 2-11.

2.3.1.1 LOCA Inputs and Assumptions

The key inputs used in this analysis are included in Table 2-12. These inputs and assumptions fall into three categories: Radionuclide Release Inputs, Radionuclide Transport Inputs, and Radionuclide Removal Inputs.

LOCA Release Inputs

The BFN TSs specify a maximum allowable primary containment leakage rate of two percent primary containment air weight per day. This leakage rate was assumed in the AST analyses. ECCS leakage was considered in accordance with the guidance from SRP 15.6.5, Appendix B (Reference 20). A five gallon per minute (gpm) leak rate into the reactor building was used starting at the onset of the event. This leakage is more than the operational ECCS leakage for BFN (approximately 1 gpm). The plant TSs total limit allowable MSIV leakage of 150 scfh (maximum of 100 scfh in any one line) was assumed by the analyses. The analyses conservatively assume no reduction in these leak rates over the 30 day duration of the dose calculation.

Primary Containment leakage via the hardened wetwell vent (HWWV) bypasses secondary containment and is released unfiltered to the atmosphere via the top of the stack. A leakage of 10 scfh is conservatively assumed to begin 8 hours after the beginning of the event. This delay is based upon the leakage rate and the large volume of the HWWV piping between the drywell and the stack.

LOCA Transport Inputs

All three trains of SGT are conservatively assumed to be in operation at the beginning of the accident. This maximizes the release from secondary containment. If only two of the SGT trains are in operation, a short time period exists at the start of the accident during which the secondary containment can become pressurized relative to the outside environment. However, negative pressure would be re-established in secondary containment prior to the gap release at two minutes specified by RG 1.183. Accordingly, three train operation of SGT is the conservative case. The reactor building pressure is negative throughout the RG 1.183 release phases and the primary containment leakage (with the exception of the MSIV leakage and the leakage through the HWWV after eight hours) is assumed to be collected by SGT and directed to the stack. A portion of the stack flow (10 scfm) is assumed to leak through the stack backdraft isolation dampers and released as a ground level release at the base of the stack. This amount of leakage is within the bounds of procedural controls.

Since the main steam lines and the main condenser are seismically-rugged, and are assumed to remain intact, the MSIV leakage eventually collects in the main condenser (except for a small portion that is assured to bypass the main condenser). The LOCA analysis also assumes that one of the four inboard MSIVs fails to close (this postulated single failure results in the worst case dose consequences). Therefore, three of the steam lines have a closed space between the inboard and outboard MSIVs. The piping volume between the outboard MSIVs and the assorted valving downstream (i.e., main turbine stop valves, main turbine bypass valves, reactor feed pump high pressure steam stop valves, etc.) also comprises a large, closed space. In each of the three steam lines that are fully isolated, a well mixed control volume is defined in the space between the closed MSIVs as well as in the space downstream of the outboard MSIVs.

Only the control volumes in the horizontal portions of this main steam piping are credited in the analyses for activity disposition. The space down stream of the MSIV in the faulted steam line (the one with only the outboard MSIV closed) is credited with an isolated control volume only in the space from the outboard MSIV to the point where the drain line pathway to the main condenser connects to the steam line. This volume is consistent with others in that it is made up of horizontal piping also.

For conservatism, a maximum MSIV leakage per line of 100 scfh is assumed to exist in the faulted line. One of the fully isolated lines is assumed to leak at 50 scfh, while the other two are assumed to be leak-tight. This set of assumptions minimizes credit for retention in the steam lines.

The pressure in the space between the closed MSIVs is assumed to be that of the containment, but the temperature is assumed to be the normal operating conditions of the steam line. In the steam line outboard of the MSIVs, the pressure is assumed to be atmospheric, the temperature is also assumed to be the normal operating. The condenser is assumed to be at standard conditions. MSIV leakage at the test pressure is converted into volumetric flow rates based upon post-LOCA drywell temperature and pressure.

The MSIV leakage from the main condenser is assumed to be released directly to the environment as a turbine building release with no credit for turbine building hold-up.

The control room would automatically isolate and the CREV is automatically initiated at the onset of the accident due to high drywell pressure or low reactor level trip signals. However, for conservatism, a 10 minute delay in CREV initiation is assumed in the dose analysis. A schematic of the transport model is provided in Figure 2-1.

LOCA Removal Inputs

LOCA activity release is partially removed by natural deposition in the drywell, natural deposition in the main steam lines and the condenser, and by removal of particulates by the SGT and CREV HEPA filters.

In the Drywell

Natural removal of activity is credited in the drywell using the 10th percentile values from the models of RADTRAD Table 2.2.2.1-3. The elemental iodine is assumed to have the same removal rate as the particulate (noting that the total surface area of the particulate is substantially greater than that of the drywell structures and that the Standard Review Plan Section 6.5.2 elemental iodine wall deposition rate (λ_w) is greater in any case). No credit is taken for organic iodine and noble gas removal.

In the Steam Lines

The AEB-98-03 (Reference 22) model is used to obtain the deposition velocity for particulate. The AEB-98-03 model assumes a well mixed control volume. Since the flow in the steam line is expected to be plug flow (because of the values of the assumed MSIV leakage), it is justified (according to AEB-98-03) to use the median value for the deposition velocity found in Appendix A of that document. The horizontal cross-section of the steam line is used as the surface area for deposition.

The RADTRAD Bixler model is used for deposition of elemental iodine. As with particulate deposition, no credit is taken for cooling of the steam lines with time and the associated increase in residence time.

For the steam lines in which both MSIVs are closed, there are two steam line control volumes in series. In the second (outboard) control volume, it would be expected that the particulate concentration and the representative deposition velocity would be lower. Therefore, the distribution of deposition velocity for particulates in the second control volume has been adjusted to reflect the faster settling particles that have already been removed in the first control volume. The median deposition velocity in the first control volume is 1.17E-3 m/s (the AEB-98-03 median value), but it is calculated to be 2.7E-4 m/s in the second control volume.

Removal In the Condenser

Particulate deposition in the main condenser is treated using the same approach as that for the steam lines. The effective volume of the main condenser (for hold-up) is based on crediting 90% of the nominal condenser volume and none of the volume of the low pressure turbine.

Since the efficiency of the condenser in removing both particulate and elemental iodine is determined by the relative removal and leakage lambdas (and since the main condenser volume is in the denominator for both), the only things determining the condenser removal efficiency are: (1) deposition velocity, (2) deposition area, and (3) volumetric flow out of the main condenser. For particulate, the sedimentation velocity in the main condenser is assumed to be the flow-weighted average of the median values exiting the two steam lines with leakage, and that flow-weighted average is 3.47E-4 m/s. The sedimentation area is the assumed effective volume of the main condenser divided by the sedimentation height of the main condenser.

The elemental iodine removal rate in the main condenser could be appropriately calculated using the 4.9 meter/hour (1.36E-3 m/s) deposition velocity from the SRP 6.5.2 for λ_w, but instead, it is assumed to be the same as particulate. This is especially conservative, because not only is the SRP 6.5.2 elemental iodine deposition velocity nearly four times greater than that for sedimentation; but also, elemental iodine deposition occurs on vertical and overhead surfaces as well as on horizontal surfaces facing upward.

Combined Efficiencies for Steam Lines and Main Condenser

The steam line and main condenser removal efficiencies for particulate and elemental iodine may be combined by weighting the steam line removal according to flow and then placing these removal efficiencies in series with that of the main condenser. These efficiencies included a condenser bypass term of 0.5 percent of the total MSIV leakage.

2.3.1.2 Main Steam Line Break Accident Inputs and Assumptions

The main steam line break accident assumes a double ended break of one main steam line outside the secondary containment with displacement of the pipe ends that permits maximum blowdown rates. The analysis also assumes isolation of the control room habitability zone and the initiation of the CREV System by the control room normal ventilation intake radiation monitors on high radiation.

The radiological consequences of the design basis main steam line break accident were analyzed using TVA's STP, COROD, and FENCDOSE codes. The evaluation of fuel performance for the main steam line break accident determined that no fuel rod failures are postulated for this event. Two cases were evaluated that corresponded to the iodine concentration in the primary coolant:

- An assumed pre-accident spike of 32 μCi/gm Dose Equivalent (DE) I-131 (conservative value based on the TS maximum allowed value of 26 μCi/gm DE I-131).
- A value of 3.2 μCi/gm DE I-131 corresponding to the maximum TS value allowed for continued operation.

The break mass released includes the line inventory plus the system mass released through the break prior to isolation. Break isolation was assumed in 5.5 seconds. This assumption is consistent with the isolation time used in evaluation of the pressure, temperature, pipe whip, and jet impingement effects for main steam line breaks outside of the drywell. This is the maximum isolation time for an MSIV given the expected 3 to 5 second isolation and includes isolation instrumentation response time. This results in the maximum radiological release for analysis. The analysis assumes an instantaneous ground level puff release.

RG 1.183, Section 4.4 of Appendix D, indicates that the iodine species released from the main steam line should be assumed to be 95 percent CsI as an aerosol, 4.85 percent elemental, and 0.15 percent organic. The main steam line break accident analysis assumes all iodine to be elemental. This difference is inconsequential for the BFN AST analysis since no credit is taken for filtration or other removal mechanisms of iodine, such as plateout, sedimentation, condensation, or decay. The key inputs used in this analysis are included in Table 2-13.

2.3.1.3 Refueling Accident Inputs and Assumptions

This postulated refueling accident involves the drop of a fuel assembly on top of the reactor core during refueling operations. The drop over the reactor core is more limiting than the drop over the spent fuel pool since the kinetic energy for the drop over the reactor core area (greater than 23 feet) produces a larger number of damaged fuel pins on impact than the shorter drops that could occur over the fuel pool.

All the refueling accident activity is assumed released to the environment from the refuel building ventilation system with no credit for reactor building holdup or dilution. Not crediting any dilution, holdup, or cleanup by SGT of the activity released from the pool represents a more conservative basis than that used in the existing licensing basis analysis.

All current fuel types are bounded by this analyses. The key inputs used in this analysis are included in Table 2-14.

2.3.1.4 Control Rod Drop Accident Inputs and Assumptions

The BFN analysis for the control rod drop accident considers the worst-case radiological exposure release path. The analysis assumes the condenser is evacuated for 30 days after the rod drop either by the steam jet air ejectors (steady state operation) or the Mechanical Vacuum Pump (MVP) operation (at startup) (Reference 23). While plant interlocks and procedures essentially prevent power operation with the MVPs in service, exhaust via the MVPs provide the greatest amount of activity released, and therefore this pathway is used for the analysis.

The activity released from the core is instantaneously released to the main condenser. From the condenser the activity flows to the stack where fumigation conditions are considered from 0 to 30 minutes. It is assumed ten scfm of leakage enters the stack room where mixing occurs prior to release to the environment at the base of the stack. Releases from the damaged fuel, and deposition in the main condenser are per Appendix C of RG 1.183. A schematic of the transport model is provided in Figure 2-2. The key inputs used in this analysis are included in Table 2-15.

2.3.2 <u>Suppression Pool pH Control</u>

NUREG-1465 notes that SRP 6.5.5 (Reference 24), allows credit for fission product scrubbing in the suppression pool. Although fission product removal by suppression pool scrubbing is not credited in the BFN analyses, natural removal by sedimentation is credited; and this will lead to a large fraction of activity being deposited in the pool water. The pool water will also retain soluble gaseous and soluble fission products such as iodides and cesium, but not noble gases. Once deposited the iodine will remain in solution as long as the suppression pool pH is maintained at or above 7.0.

It is expected that the initial effects on post-accident suppression pool pH will come from rapid fission product transport and the formation of cesium compounds, which would result in increasing the suppression pool pH. However, cesium compounds are not credited in the long-term pH analyses and the determination of the final (30 day) pH value. As radiolytic production of nitric acid and hydrochloric acid proceeds, and these acids are transported to the pool over the first days of the event, the pH would become more acidic.

Upon detection of high drywell radiation associated with the postulated activity release, plant procedures will be revised to require manual initiation of SLC injection. The buffering effect of SLC injection within several hours is sufficient to offset the effects of these acids that are transported to the pool and maintain suppression pool pH at or above 7.0.

The current design function of the SLC System is to provide a backup method, independent of control rods, to make the reactor subcritical over a full range of operating conditions. The system actuation requirements for reactivity control are explicitly addressed in the BFN Emergency Operating Instructions (EOIs). The SLC system is designed as a seismic Class I system for withstanding specified earthquake loadings. Additionally, the SLC System pumps, valves, and controls are powered from the diesel generator in absence of normal power. The current TS requires the system be maintained in an operable status whenever the reactor is in modes 1 or 2.

The SLC System is currently classified as a special safety system as defined in UFSAR Section 1.2. The SLC System will also be credited for limiting radiological dose following LOCAs involving fuel damage in accordance with the AST analyses for suppression pool pH control; however, the system will remain classified as a special safety system instead of being classified as a safety system.

A core damage event large enough to release substantial quantities of fission products into the drywell will result in high drywell radiation alarms. The operational response procedures will be revised to include instructions to manually actuate the SLC System injection. The AST analysis provides for SLC System actuation at 2 hours of accident initiation and completion of injection of an adequate volume and content of SPB within several hours, which will ensure the suppression pool pH remains at or above 7.0 for 30 days.

Initiation of the SLC System following fuel damage to control suppression pool pH is a new operator action during a DBA LOCA response. High radiation indicative of fuel failure would be sensed by two radiation monitors in the drywell and two radiation monitors in the pressure suppression chamber. Upon reaching a high radiation level, the "Drywell/Suppr Chamber Radiation High" annunciator on Panel 9-7 in the main control room would alert the operator to the fuel damage. The Alarm Response Procedure (ARP) will direct the operator to initiate SLC System injection based on the high radiation level.

Initiation of the SLC System will be accomplished from the main control room with a simple keylock switch manipulation. This switch is located on control room panel 9-5 and actuation of this switch is the only action necessary to initiate injection of the sodium pentaborate into the reactor vessel. The new SLC System function to control suppression pool pH does not involve any change to the actions needed to be performed to initiate SLC system injection. Indication of proper SLC System operation is provided in the control room as described in UFSAR Section 3.8.

During this postulated event, plant operators will be responding to the event as directed by the plant Emergency Operating Instructions (EOI). Adequate time is available for SLC System initiation during these events. Immediate initiation of the SLC System is not vital since the analysis allows for two hours before initiation. Operators are familiar with operation of the SLC System due to previous training for Anticipated Transients Without Scram (ATWS) events. Training on this new operator action will also be provided to the operators.

With certain post LOCA conditions, existing BFN procedures direct the operations of systems to accomplish a total floodup of the primary containment. This floodup uses the Ultimate Heat Sink (Tennessee River) as the preferential source of makeup water since it is the only safety related makeup water source. A review of a previous ten years of data reflect that the minimum river pH has been above a pH of 7.0, with the exception of one data point, over this time period. Although the condensate storage tank (CST) could be used as a makeup source, it is not safety related and does not have sufficient volume to flood containment without repeated refilling or the use of additional CSTs. Consequently, the addition of a large amount of water from the UHS to the suppression pool and containment inventory will not result in a pH below 7.0.

2.3.3 Main Steam Line Break Accident Puff Release Dispersion Factor

A new control room X/Q value for an instantaneous ground level puff release to the atmosphere was determined for use in the main steam line break accident radiological dose analysis. The inputs used in the determination of the X/Q value are provided in Table 2-17.

2.3.4 NUREG-0737 Evaluation

The inputs and assumptions utilized in the NUREG-0737 evaluation include the AST plant-specific fission products inventories and other applicable inputs as described in Section 2.3.1.

Table 2-1
Computer Codes Used in AST
Design Basis Accident Analyses

Task	Computer Code	Version or Revision	Comments
Used to generate the existing X/Q's	ARCON96	1996	See Note[1] NUREG/CR - 6331, Rev. 1, May 1997 NRC Sponsored
Used to determine control room operator doses for main steam line break accident	COROD	R5	TVA Code See Note[1]
Used to determine the offsite doses for main steam line break accident	FENCDOSE	R4	TVA Code See Note[1]
Used to model complex systems that take into account radioactive decay and production of daughter isotopes. Output can be in activity levels or gamma spectra.	Source Transport Program (STP)	R6	TVA Code See Note[1]
Point Kernel Integration code used for general purpose gamma shielding analysis.	MicroShield	5.03	Code used in nuclear radiological analyses. Grove Engineering. Used in safety-related applications by many nuclear plants in the U.S.
Used to calculate fission product inventories	ORIGEN	ORIGEN2 (GE) SAS2H/ ORIGEN-S	The codes are either referenced by RG 1.183 or consistent with NRC recommendation. ORNL/TM-7175 NUREG/CR-0200R6
Used to develop photon spectrum for Main Steam Line Break Accident	QADISOTP	R1	TVA Code used in nuclear radiological analyses

[1] Results were reviewed and approved by NRC in Safety Evaluation for Amendment Nos. 263 and 223 for BFN Units 2 and 3.

Table 2-1 Computer Codes Used in AST Design Basis Accident Analyses			
Task	**Computer Code**	**Version or Revision**	**Comments**
Used to determine the direct gamma shine dose due to the released isotopes in the turbine building for the Main Steam Line Break Accident	QAD-P5Z	R6	TVA Point Kernel Code used in nuclear radiological analyses.
Used for the LOCA and Control Rod Drop Accident Dose Calculations	RADTRAD	3.02a	Referenced by RG 1.183 NUREG/CR-6604 USNRC April 1998
Used to perform independent check of LOCA and Control Rod Drop Accident.	STARDOSE	03/01/1997	Polestar Applied Technology code NUREG/CR-5106
Used to evaluate Suppression Pool Water pH as a function of time	STARpH	1.04	Utilized in other AST Submittals & Developed by Polestar. NRC reviewed and approved for use of STARpH for Hope Creek. (Reference 19)
Used to perform a independent check of MicroShield.	QADMODE	Version 5.03	Point Kernel Gamma-Ray Shielding Code with Geometric Progression Building Factors

Table 2-2
Fission Product Inventory

Isotope	Ci/MWt t = 0	Ci/MWt t = 24 hr	Isotope	Ci/MWt t = 0	Ci/MWt t = 24 hr
Co58	1.430E+02	1.416E+02	Xe131M	3.544E+02	3.487E+02
Co60	1.425E+02	1.424E+02	Te132	3.829E+04	3.089E+04
Kr83M	3.432E+03	1.387E+01	I132	3.885E+04	3.184E+04
Kr85	3.601E+02	3.601E+02	I133	5.534E+04	2.559E+04
Kr85M	7.329E+03	1.811E+02	Xe133	5.504E+04	5.303E+04
Rb86	6.372E+01	6.141E+01	Xe133M	1.734E+03	1.562E+03
Kr87	1.446E+04	3.051E-02	I134	6.141E+04	1.450E-03
Kr88	2.009E+04	5.743E+01	Cs134	5.703E+03	5.697E+03
Kr89	2.521E+04	0.000E+00	I135	5.250E+04	4.189E+03
Sr89	2.786E+04	2.748E+04	Xe135	1.971E+04	1.429E+04
Sr90	3.165E+03	3.165E+03	Xe135M	1.135E+04	6.823E+02
Y90	3.283E+03	3.273E+03	Cs136	1.941E+03	1.841E+03
Sr91	3.487E+04	6.103E+03	Xe137	5.023E+04	0.000E+00
Y91	3.583E+04	3.564E+04	Cs137	4.037E+03	4.037E+03
Sr92	3.677E+04	7.922E+01	Ba137M	3.829E+03	3.810E+03
Y92	3.696E+04	1.168E+03	Xe138	4.757E+04	1.172E-26
Y93	4.147E+04	8.084E+03	Ba139	4.930E+04	4.170E-01
Zr95	4.880E+04	4.822E+04	Ba140	4.909E+04	4.644E+04
Nb95	4.897E+04	4.897E+04	La140	5.231E+04	5.079E+04
Zr97	4.953E+04	1.851E+04	La141	4.498E+04	7.085E+02
Mo99	5.088E+04	3.956E+04	Ce141	4.535E+04	4.463E+04
Tc99M	4.454E+04	3.772E+04	La142	4.397E+04	1.035E+00
Ru103	4.094E+04	4.018E+04	Ce143	4.245E+04	2.597E+04
Ru105	2.710E+04	6.615E+02	Pr143	4.113E+04	4.075E+04
RH105	2.559E+04	1.840E+04	Ce144	3.810E+04	3.810E+04
Ru106	1.488E+04	1.486E+04	Nd147	1.806E+04	1.698E+04
Sb127	2.796E+03	2.369E+03	Np239	5.201E+05	3.902E+05
Te127	2.773E+03	2.580E+03	Pu238	2.805E+02	2.805E+02
Te127M	3.721E+02	3.719E+02	Pu239	1.234E+01	1.238E+01
Sb129	8.457E+03	1.952E+02	Pu240	1.730E+01	1.730E+01
Te129	8.326E+03	1.236E+03	Pu241	4.450E+03	4.448E+03
Te129M	1.615E+03	1.590E+03	Am241	5.449E+00	5.470E+00
Te131M	5.155E+03	2.976E+03	Cm242	1.234E+03	1.234E+03
I131	2.669E+04	2.481E+04	Cm244	5.697E+01	5.697E+01

Table 2-3 X/Q Values for Radiological Dose Calculations Top of Stack Releases (LOCA and Control Rod Drop Accident)				
Time Period	Control Room (sec/m^3)		EAB(2)	LPZ
	Unit 1 Intake	Unit 3 Intake	(sec/m^3)	(sec/m^3)
Fumigation	3.40E-5	3.02E-5	2.35E-5[1]	1.26E-5
0-2 hrs	9.08E-13	1.41E-7	1.19E-6[1]	1.13E-6
2-8 hrs	3.41E-13	4.50E-8	—	5.75E-7
8-24 hrs	2.09E-13	2.54E-8	—	4.10E-7
1-4 days	7.21E-14	7.36E-9	—	1.97E-7
4-30 days	1.57E-14	1.24E-9	—	6.88E-8

1 These values were incorrectly listed in Reference 14; however, the correct values were used as the basis of Reference 14.

2 Maximum EAB TEDE for any 2 hour period.

Table 2-4 X/Q Values for Radiological Dose Calculations Base of Stack Releases (LOCA and Control Rod Drop Accident)				
Time Period	Control Room (sec/m^3)		EAB(2)	LPZ
	Unit 1 Intake	Unit 3 Intake	(sec/m^3)	(sec/m^3)
0-2 hrs	2.00E-4	8.60E-5	2.62E-4	1.31E-4
2-8 hrs	1.28E-4	6.46E-5	—	6.61E-5
8-24 hrs	5.72E-5	2.80E-5	—	4.69E-5
1-4 days	4.05E-5	2.00E-5	—	2.23E-5[1]
4-30 days	3.09E-5	1.53E-5	—	7.96E-6

1 Typo in Reference 14; same as value for turbine building release.

2 Maximum EAB TEDE for any 2 hour period

Table 2-5
X/Q Values for Radiological Dose Calculations
Refueling Vent Releases

(Refueling Accident Only)

Time Period	Control Room (sec/m^3)		EAB	LPZ
	Unit 1 Intake	Unit 3 Intake	(sec/m^3)	(sec/m^3)
0-2 hrs	4.60E-4	*	2.62E-4	1.31E-4

*Bounded by the Unit 1 Intake

Table 2-6
X/Q Values for Radiological Dose Calculations
Turbine Building Exhaust Release
(Main Steam Line Break Accident Only)

Time Period	EAB (sec/m^3)	LPZ (sec/m^3)
0-2 hrs	2.62E-4	1.31E-4

Table 2-7
X/Q Values for Radiological Dose Calculations
Turbine Building Roof Ventilator Releases

(Post LOCA MSIV Leakage Only)

Time Period	Control Room (sec/m^3)		EAB[(1)]	LPZ
	Unit 1 Intake	Unit 3 Intake	(sec/m^3)	(sec/m^3)
0-2 hrs	*	2.17E-4	2.62E-4	1.31E-4
2-8 hrs	*	1.64E-4	—	6.61E-5
8-24 hrs	*	7.89E-5	—	4.69E-5
1-4 days	*	4.33E-5	—	2.23E-5
4-30 days	*	3.35E-5	—	7.96E-6

*Bounded by the Unit 3 Intake

1 Maximum EAB TEDE for any 2 hour period

Table 2-8 Fuel Data			
Fuel Data	**General Electric**	**Framatome**	**Framatome**
Fuel Type	GE14	A10	BLEU
Initial Bundle Mass of Uranium (kg)	182.0	177.7	177.7
Initial Core Average Enrichment (U-235 wt%)	4.6	4.5	4.95
Core Average Bundle Power (MWt/bundle)	5.28	5.28	5.28
End of Cycle Core Average Exposure (GWd/MT)	35.0	37.0	37.0

Table 2-9 LOCA Release Fractions as Release Rates Over the Duration		
Time Period (seconds)	Fraction of core inventory	
0 - 120	No Release	
120 - 1920	Gases	Xe, Kr – 0.1/hr (0.05 total) Elemental I – 4.9E-3/hr (2.4E-3 total) Organic I – 1.5E-4/hr (7.5E-5 total)
	Aerosols	I, Br – 0.095/hr (0.0475 total) Cs, Rb – 0.1/hr (0.05 total)
1920 - 7320	Gases	Xe, Kr – 0.63/hr (0.95 total) Elemental I – 8.1E-3/hr (1.2E-2 total) Organic I – 2.5E-4/hr (3.8E-4 total)
	Aerosols	I, Br – 0.158/hr (0.2375 total) Cs, Rb – 0.133/hr (0.2 total) Te Group – 0.033/hr (0.05 total) Ba, Sr – 0.013/hr (0.02 total) Noble Metals – 1.7E-3/hr (2.5E-3 total) La Group – 1.3E-4/hr (2E-4 total) Ce Group – 3.3E-4/hr (5E-4 total)

Table 2-10
CREV/SGT Functions Modeled in Dose Analyses

DBA Dose Analysis	CREV			SGT		
	Pressurization Mode	HEPA Particulate Removal	Charcoal Adsorber	Flow/ Secondary Containment	HEPA Particulate Removal	Charcoal Adsorber
LOCA	Y	Y	N	Y	Y	N
Main Steam Line Break Accident	Y	N[1]	N	N[2]	N[2]	N
Refueling Accident	N	N[1]	N	N[3]	N[3]	N
Control Rod Drop Accident	Y	N	N	N[2]	N[2]	N

1 No particulates are released to the atmosphere; therefore no particulate filtering is necessary in analysis.
2 No release to secondary containment.
3 No credit taken for holdup or filtering in secondary containment.

Table 2-11 Accident Radiological Consequence Analyses Inputs	
Input/Assumption	**Value**
CREV Intake Flow Rate	6717 scfm
CREV Makeup Filtered Flow Rate	3000 scfm
CREV Unfiltered Inleakage Rate	3717 scfm
CREV HEPA Filter Efficiency	90% Particulate
CREV Charcoal Adsorption Efficiency	No credit taken
Control Room volume	210,000 ft^3
SGT Flow Rate	24,750 scfm
SGT HEPA Filter Efficiency	90% Particulate
SGT Charcoal Adsorption Efficiency	No credit taken
Environment Breathing Rate	0-8 hours: 3.5E-04 m^3/sec 8-24 hours: 1.8E-04 m^3/sec 1-30 days: 2.3E-04 m^3/sec
Control Room Breathing Rate	3.5E-04 m^3/sec
Control Room Occupancy Factors	0-1 day: 1.0 1-4 days: 0.6 4-30 days: 0.4

Table 2-12 LOCA Inputs	
Input/Assumption	**Value**
Fission Products Release Fractions	Regulatory Guide 1.183 Table 1 **BWR Core Inventory Fraction Released Into Containment** **Group** / **Gap Release Phase** / **Early In-vessel Phase** / **Total** Noble Gases 0.05 0.95 1.0 Halogens 0.05 0.25 0.3 Alkali Metals 0.05 0.20 0.25 Tellurium Metals 0.00 0.05 0.05 Ba, Sr 0.00 0.02 0.02 Noble Metals 0.00 0.0025 0.0025 Cerium Group 0.00 0.005 0.0005 Lanthanides 0.00 0.005 0.0002
Fission Product Release Timing	Regulatory Guide 1.183 Table 4 **LOCA Release Phases BWR** Phase / Onset / Duration Gap release 2 min 0.5 hr Early In-Vessel 0.5 hr 1.5 hr
Fission Product Iodine Chemical Form	Particulate 95% Elemental 4.85% Organic 0.15%
Control Room Isolation/CREV Initiation	10 minutes
ECCS Leakage Release Fractions	Ten percent of the radioiodine in the leaked coolant is assumed to become airborne in the reactor building (secondary containment). Of this activity, 97% is assumed to be elemental iodine and 3% is assumed to be organic iodine.
Flow Rates	
Primary Containment Leak Rate (30 days)	2 % containment air weight/day
Secondary Containment Bypass Leak Rate (30 Days)	HWWV = 10 scfh beginning at t>8 hours
Assumed ECCS Leak Rate (30 days)	5 gpm
ECCS Leakage Temperature	<212°F

Table 2-12 LOCA Inputs	
Input/Assumption	**Value**
MSIV Leak Rate at test pressure of 25 psig	150 scfh total 100 scfh maximum for one line
Leakage at base of stack (stack bypass)	10 scfm
MSIV Leakage that Bypasses Main Condenser	0.5% (percentage of total MSIV leakage)
CAD vent rate	139 scfm for 24 hrs @ 10 days, 20 days, 29 days
Volumes	
Drywell Airspace	159,000 ft^3 (Min value used for dose calculation)
Torus Airspace	119,400 ft^3 (Minimum)
Suppression Pool	121,500 ft^3 (Minimum)
Reactor Building Free Volume	1,931,502 ft^3 (50% of this value used due to incomplete mixing)
Stack Room	69,120 ft^3 (50% of this value used due to incomplete mixing)
High Pressure Turbine	568.6 ft^3 (No credit taken)
Low Pressure Turbine	51,000 ft^3 (No credit taken)
Removal Inputs	
Drywell Natural Deposition	Particulate: Power's Model, 10th percentile values(conservative compared to SRP 6.5.2 λ_w. Elemental: Same as particulate.
Drywell Accident Conditions (maximum)	P = 48.3 psig, T = 294.9 Degrees F
Surface Area for Elemental Iodine Deposition in Drywell	3409 m^2

Table 2-12 LOCA Inputs		
Input/Assumption Steam Line and Main Condenser Removal Efficiencies:	**Value**	
Condenser Volume	90 Percent of 136,000 ft^3 or 122,400 ft^3	
Steam Line Conditions	Saturated Conditions at 1050 psia	
Steam Line Volume: Inboard to Outboard MSIV	53.7 ft^3	
Steam Line Volume: Outboard MSIV to drain line	173.1 ft^3	
Sedimentation Height	27.2 ft	
	Removal Efficiency for Aerosol Particles	Removal Efficiency for Elemental Iodine
Steam Line Leakage (Drywell to Main Condenser) (These removal efficiencies applied to a leakage entering the main condenser volume include removal in the condenser downstream)	99.87%	99.01%
Main Condenser Bypass (Drywell to Environment)	89.33%	16.37%

Table 2-13 Main Steam Line Break Accident Inputs	
Input/Assumption	**Value**
Mass Release	11,975 lbm steam 42,215 lbm water (saturated @ 898psia)
MSIV Isolation Time	5.5 seconds
DE I-131 Equilibrium Value	3.2 μCi/gm
DE-I-131 Pre-Accident Spike	32 μCi/gm (Conservative to TS value of 26μCi/gm)
Iodine Species Release Fraction	All Assumed Elemental

Table 2-14 Refueling Accident Inputs	
Input/Assumption	**Value**
Number of Failed Rods	111
Radial Peaking Factor	1.5
Fuel Decay Period	24 hours
Pool Water Iodine Decontamination Factor Elemental Organic	 500 1
Release Period	Instantaneous
Reactor Building Ground Release Location	Reactor Building Refueling Zone Vent (No credit for holdup or SGT operation)
Release Fractions	Noble Gases excluding Kr-85 5 percent Kr-85 10 percent I-131 8 percent Iodines except I-131 5 percent

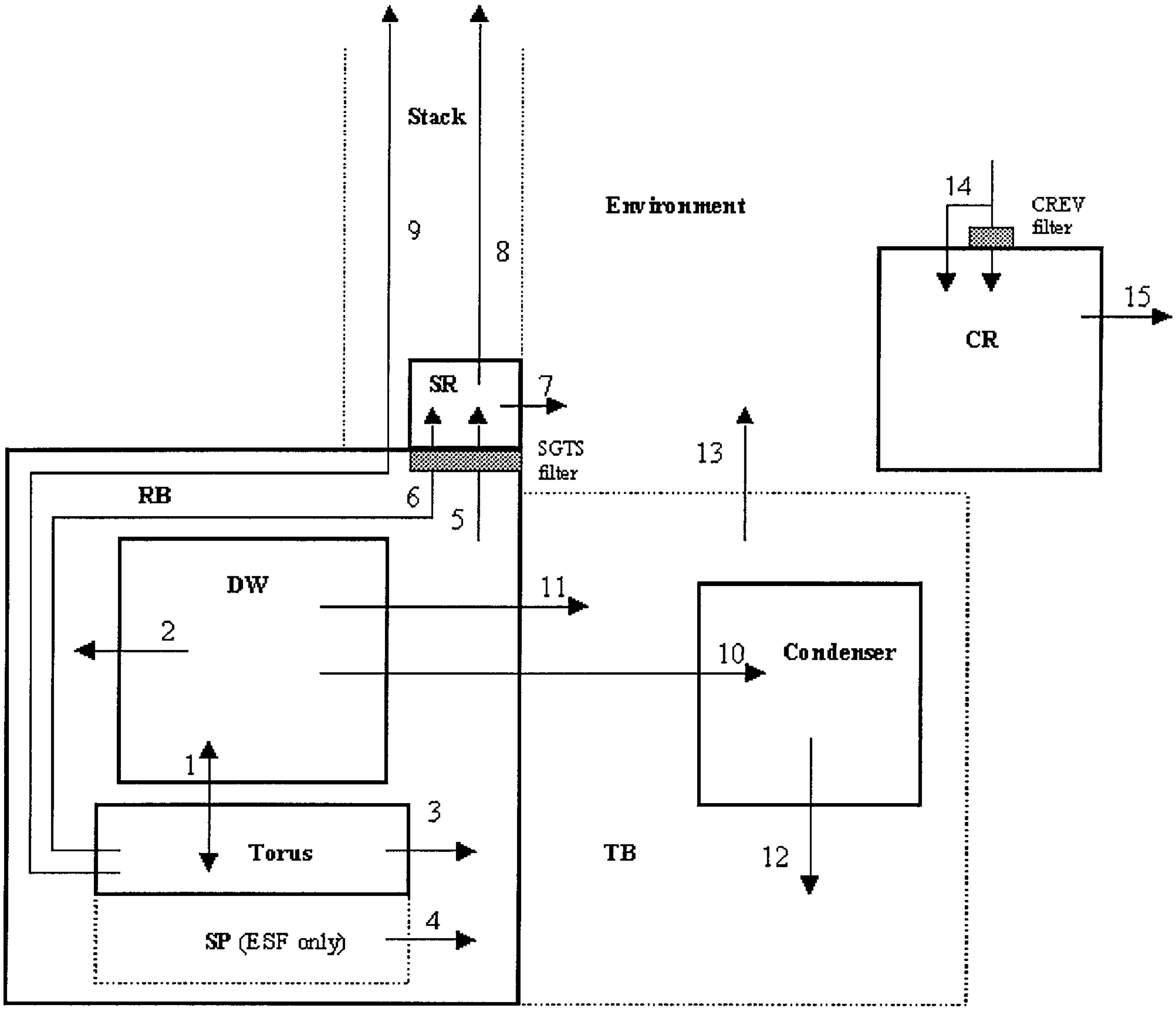
Table 2-15 Control Rod Drop Accident Inputs	
Input/Assumption	**Value**
Number of Failed Rods	850
Percent Fuel Melt for Failed Rods	0.77 %
Radial Peaking Factor	1.50
Release Period	24 hours
Main Condenser and Low Pressure Turbine Free Volume	187,000 ft^3
Stack Room Volume	69,120 ft^3 (50% of this value used due to incomplete mixing)
Assumed Base of Stack Leakage	10 cfm
Mechanical Vacuum Pump Flowrate	1850 scfm @ 7" Hg
Gap Release Fractions	Noble Gas 10% Iodine 10% Br 5% Cs, Rb 12% Te Group 0% Ba, Sr 0% Noble Mtls 0% Ce Group 0% La Group 0%
Core Melt Release Fractions	Noble Gas 100% Iodine 50% Br 30% Cs, Rb 25% Te Group 5% Ba, Sr 2% Noble Mtls 0.25% Ce Group 0.05% La Group 0.02%
Activity that reaches the condenser	Noble Gas 100% Iodine 10% Br 1% Cs, Rb 1% Te Group 1% Ba, Sr 1% Noble Mtls 1% Ce Group 1% La Group 1%

Table 2-15 Control Rod Drop Accident Inputs	
Input/Assumption	**Value**
Activity released from the condenser	Noble Gas 100% Iodine 10% Br 1% Cs, Rb 1% Te Group 1% Ba, Sr 1% Noble Mtls 1% Ce Group 1% La Group 1%

Table 2-16 Suppression Pool pH Control Inputs	
Input/Assumption	**Value**
Maximum Suppression Pool Volume	131,400 ft^3
Containment Free Volume	278,400 ft^3
Reactor Coolant System Inventory	1.226E-06 lbm
Sodium Pentaborate Injectable Volume	4000 gal
SLC ($Na_2O*5B_2O_3*10H_2O$) injected	8 weight percent
Sodium Pentaborate Enrichment	62.9 mole% B10
Initial Suppression Pool pH	5.3
Average suppression pool temperature	132°F
Mass of Polyvinyl Chloride Jacket in the Drywell	2865 lbm
Mass of Hypalon Jacket in the Drywell	868 lbm
Average Cable Outside Diameter	0.89 inches
Average Cable Jacket Thickness	72 mils
Percent of Drywell Cable in Conduit	30%
Conduit Material	Aluminum
Conduit wall thickness	0.1 inch
Conduit air gap	0.25 inch

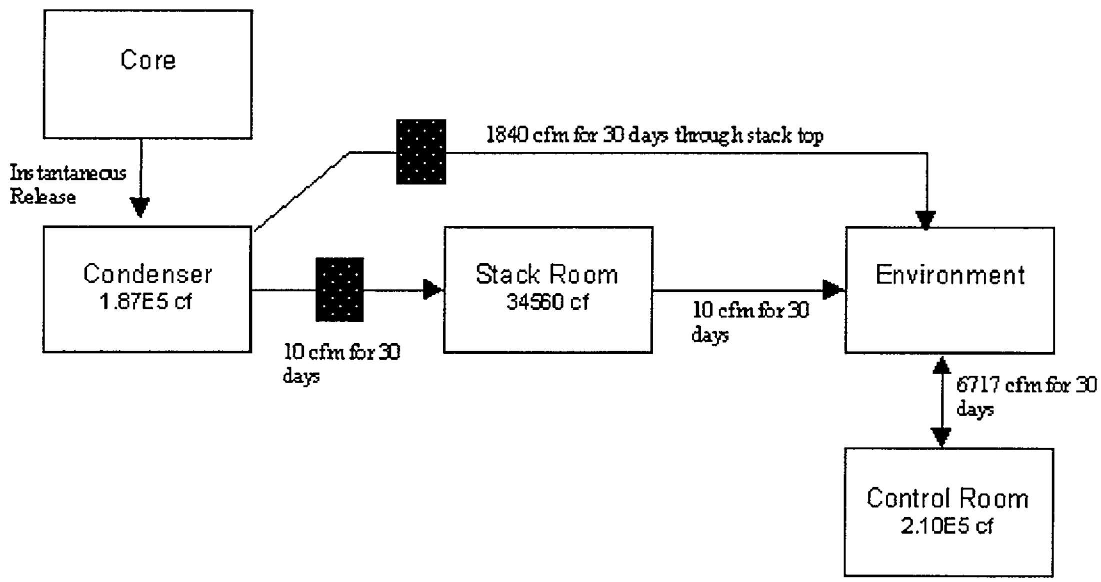
Table 2-17 Main Steam Line Break Accident Puff Release X/Q Inputs	
Input/Assumption	**Value**
Mass Release	11,975 lbm steam 42,215 lbm water (saturated @ 898psia) Assumed instantaneous release
Bubble Geometry	Spherical & Hemispherical Cases Considered
Turbine Building Perimeter Dimension	~1500 ft

Figure 2-1: LOCA Transport Model



Flow-path	Description
1	Drywell/Torus Mixing (After Release)
2/3	Primary containment leakage
4	ECCS Leakage
5	SGT Flow
6	CAD venting
7	Base of stack release (stack bypass)
8	Stack release
9	HWWV Leakage
10	MSIV Leakage
11	MSIV Leakage - Condenser bypass
12	Condenser leakage
13	No credit taken for holdup in the Turbine Building
14	CREV filtered/unfiltered intake
15	CREV exhaust

Figure 2-2: Control Rod Drop Accident Transport Model



3. RESULTS

3.1 Evaluation Results

3.1.1 Accident Radiological Consequence Analyses

The postulated accident radiological consequence analyses were updated for AST implementation impact. Comparison of updated AST doses to existing licensing basis doses considers impact from the assumed operation at EPU conditions (4031 MWt (102% of 3952 MWt)) as well as the change in analysis methodology.

3.1.1.1 LOCA

The radiological consequences of the DBA LOCA were analyzed using the RADTRAD code and the inputs/assumptions defined in Section 2.3.1.1 of this report. The post accident doses are the result of the following activity considerations:

1. Primary to secondary containment leakage. This leakage is directly released into secondary containment and filtered by SGT System prior to elevated release through the plant stack with stack bypass released at ground level. No credit is taken for charcoal adsorber action.

2. ECCS leakage into the secondary containment. This leakage is directly released into the secondary containment environment and the airborne portion is filtered by SGT System prior to elevated release through the plant stack with stack bypass released at ground level. No credit is taken for charcoal adsorber.

3. MSIV leakage from the primary containment into the main condenser (with a fraction that bypasses the main condenser directly to the atmosphere). Leakage passes through the alternate MSIV leakage pathway to the main condenser with credit for deposition before it is released, undiluted and unfiltered, through the turbine building vents.

4. HWWV leakage from primary containment. This leakage is directly released (after a eight hour delay) to an elevated release through the plant stack.

5. Post-DBA LOCA radiation shine dose to personnel within the control room from activity released to the reactor building and from activity contained in Core Spray System piping.

The EAB, LPZ, and control room calculated doses are within the regulatory limits. Table 3-1 presents the results of the LOCA radiological consequence analysis.

3.1.1.2 Main Steam Line Break Accident

The EAB, LPZ and control room calculated doses are within the regulatory limits for the cases analyzed. The control room doses were determined using the new X/Q value for the instantaneous puff release. Table 3-2 presents the results of the main steam line break accident radiological consequence analysis.

3.1.1.3 Refueling Accident

The radiological consequences of the design basis refueling accident were analyzed using a simplified configuration of one unique release pathway using the turbine building exhaust release X/Q for the EAB and LPZ, and the refueling X/Q for the control room along with the inputs/assumptions defined in Section 2.3.1.3 of this report. The EAB, LPZ, and control room calculated doses are within the regulatory limits. Table 3-3 presents the results of the refueling accident radiological consequence analysis.

3.1.1.4 Control Rod Drop Accident

The radiological consequences of the design basis control rod drop accident were analyzed using the RADTRAD code and the inputs/assumptions defined in Section 2.3.1.4 of this report. The EAB, LPZ, and control room calculated doses are within the regulatory limits. Table 3-4 presents the results of the control rod drop accident analysis.

3.1.2 <u>Suppression Pool pH Control</u>

The re-evolution of elemental iodine from the suppression pool is strongly dependent on suppression pool pH. The analysis assumed that SPB was injected via SLC within several hours of the onset of a LOCA. The conservative modeling of the primary containment cabling results in the production of a large amount of hydrochloric acid. The minimum suppression pool pH at 30 days post-LOCA remains above 7.0, which satisfies the conditions for inhibiting the release of the chemical form of elemental iodine in the elemental form from the suppression pool water. The suppression pool pH response over time is shown in Figure 3-1.

The quantity of SLC calculated as necessary to meet AST requirements is above the current TS requirements; therefore, TS revisions are proposed which increase the quantity of SLC required. Based on these TS changes, AST analysis for suppression pool pH control, the SLC system will be credited for limiting radiological dose following LOCAs involving fuel damage.

3.1.3 Main Steam Line Break Accident Instantaneous Ground Level Puff Release Dispersion Factor

The new control room X/Q value for an instantaneous ground level puff release to the atmosphere was calculated for use in the main steam line break accident radiological dose analysis. The X/Q value is shown in Table 3-5.

3.1.4 NUREG-0737 Evaluation

The results of the NUREG-0737 evaluation are summarized below.

- Post-Accident Vital Area Access and Sampling - The results of the revision of post-accident mission doses demonstrate that the current calculated doses (based on TID-14844 source terms) bound the doses that would be calculated based on AST source terms. The evaluated mission doses for BFN remain less than 5 rem TEDE.

- Post-Accident Radiation Monitor - The containment high range radiation monitors used to monitor post-accident primary containment radiation levels were evaluated for the impact of AST. The monitors continue to provide their design function and envelop the projected radiation exposure rates.

- Control Room Radiation Protection - The resultant doses to the control room for each of the four DBAs analyzed for AST have been determined. The results of these analyses are presented in Section 3.1.1.

- Radioactive Sources Outside the Primary Containment - The contribution of radiological dose consequences as a result of core spray piping shine and ECCS leakage was determined as part of the radiological dose analysis for the LOCA. The results of this analysis are presented in Section 3.1.1.1.

3.2 Summary

Implementation of the AST as the plant radiological consequence analyses licensing basis requires a license amendment pursuant to the requirements of 10 CFR 50.67. Radiological dose analyses were performed for the four DBAs with a potential for offsite/control room dose. Doses calculated with the AST for accidents involving damaged fuel reflect delayed and/or reduced activity releases (relative to those of TID-14844 and RG 1.3) to the primary containment, reactor building, and or/or steam lines, as applicable. Offsite and control room doses remain within regulatory requirements.

Table 3-1
LOCA Radiological Consequence Analysis

(rem TEDE)

Dose Component	**Offsite Dose**		**Control Room Dose**
	EAB	**LPZ**	
Base of Stack	—	1.08E-2	4.49E-3
Top of Stack	—	5.68E-1	2.43E-1
Turbine Building Roof	—	3.02E-1	1.13E-1
ECCS Leakage - Base of Stack	—	1.25E-2	1.21E-2
ECCS Leakage - Top of Stack	—	3.52E-1	1.12E-1
Shine	—	N/A	7.62E-1
TOTAL	**1.02**	**1.25**	**1.25**
Regulatory Limit	25	25	5
Current Analysis (Regulatory Limit) - rem	1.67E-01 (25) Gamma 1.01E-01 (300) Beta 5.84 (300) Thyroid	4.82E-01 (25) Gamma 4.84E-01 (300) Beta 8.6 (300) Thyroid	6.83E-01 (5) Gamma 1.58E-01 (30) Beta 2.95E+01 (30) Thyroid

Table 3-2
Main Steam Line Break Accident Radiological Consequence Analysis
(rem TEDE)

Case	Offsite Dose		Control Room Dose
	EAB	LPZ	
3.2 μCi/gm DE I-131	1.30E-1	6.52E-2	4.09E-2
32 μCi/gm DE I-131	1.30	6.52E-1	4.09E-1
Regulatory Limit	25	25	5
Current Analysis (Regulatory Limit) - rem[1]	3.72E-01 (25) Gamma 1.56E-01 (300) Beta 2.99E+01 (300) Thyroid	1.86E-01 (25) Gamma 7.80E-02 (300) Beta 1.49E+01 (300) Thyroid	5.30E-02 (5) Gamma 3.27E-02 (30) Beta 1.05E+01 (30) Thyroid

Table 3-3
Refueling Accident Radiological Consequence Analysis
(rem TEDE)

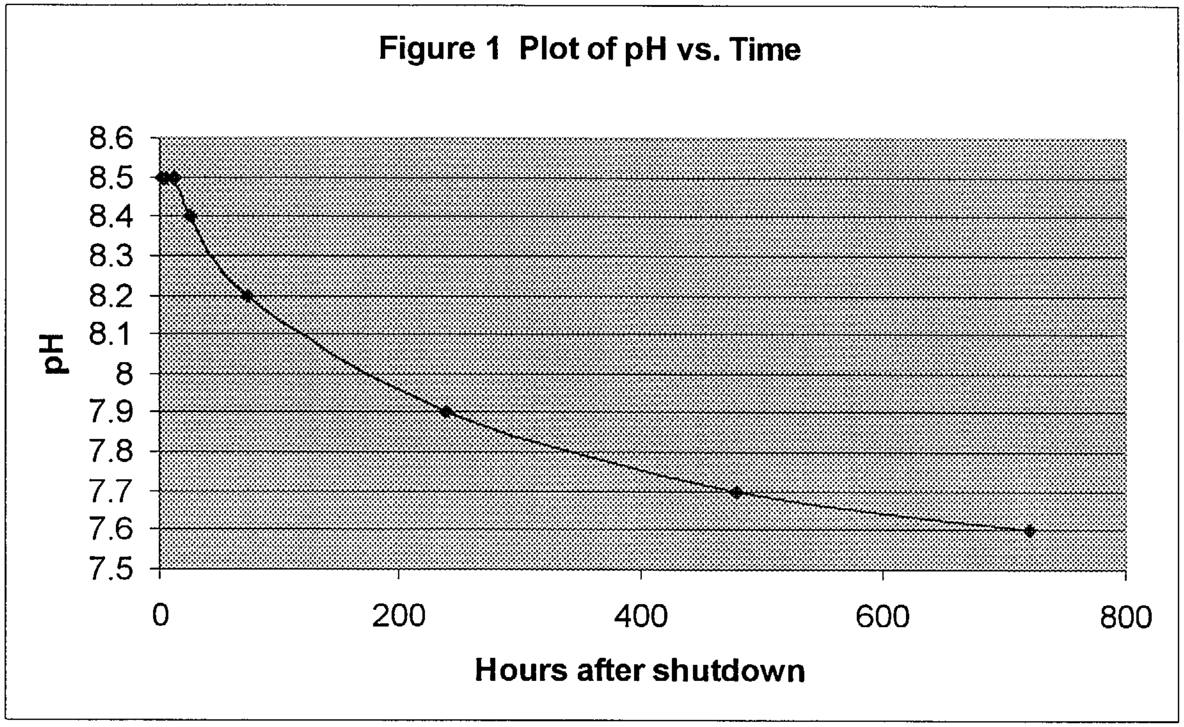
Case	Offsite Dose		Control Room Dose
	EAB	LPZ	
24 Hours after shutdown	6.70E-01	3.30E-01	3.80E-01
Regulatory Limit	6.30	6.30	5
Current Analysis (Regulatory Limit) - rem	3.37E-01 (25) Gamma 5.77E-01 (300) Beta 3.32E+01 (300) Thyroid	1.68E-01 (25) Gamma 2.89E-01 (300) Beta 1.66E+01 (300) Thyroid	4.94E-02 (5) Gamma 4.96E-01 (30) Beta 1.74 (30) Thyroid

[1] Current analysis are based on 32 μCi/gm DE I-131 limit.

Table 3-4 Control Rod Drop Accident Radiological Consequence Analysis (rem TEDE)			
Case	Offsite Dose		Control Room Dose
	EAB	LPZ	
Power Operation	1.19	6.82E-01	2.48E-01
Regulatory Limit	6.30	6.30	5
Current Analysis (Regulatory Limit) - rem	1.52 (25) Gamma 1.07 (300) Beta 1.58E+01 (300) Thyroid	8.58E-01 (25) Gamma 6.04E-01 (300) Beta 1.58E+01 (300) Thyroid	3.86E-02 (5) Gamma 4.32E-01 (30) Beta 6.3 (30) Thyroid

Table 3-5 Main Steam Line Break Accident Instantaneous Ground Level Puff Release X/Q VALUE (Main Steam Line Break Accident Only)	
Time Period	Control Room (sec/m^3)
46 secs	4.60E-4

Figure 3-1: Suppression Pool pH Response



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7. NUREG/CR-6604, RADTRAD Computer Code,: "A simplified model for Radionuclide Transport and Removal And Dose Estimation," Dated April 1998 and Supplement 1, Dated June 8, 1999.

8. STARDOSE Model report, Polestar Applied Technology, Inc., Dated January 31, 1997.

9. TVA Letter to NRC Dated September 8, 1999, Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Technical Specification (TS) Change 399 - Increased Main Steam Isolation Valve (MSIV) Leakage Rate Limits And Exemption From 10 CFR Appendix J.

10. TVA Letter Dated February 4, 2000, Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Response to Request for Additional Information Regarding Technical Specifications Change 399 - Increased Main Steam Isolation Valve (MSIV) Leakage Rate Limits and Exemption From 10 CFR 50 Appendix J - Revised TS Pages for Increased MSIV leakage Limits.

11. NRC NUREG - 6331, "Atmospheric Dispersion Relative Concentrations in Building Wakes," Revision 1, May 1997, ARCON 96, RSICC Computer Code Collection No. CCC-664.

12. NRC Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases From Light-Water-Reactors," Dated March 1, 1996.

13. NRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments of Nuclear Power Plants," Revision 2.

14. NRC Letter to TVA dated March 14, 2000, Browns Ferry Nuclear Plant, Units 2 and 3 - Issuance of Amendments Regarding Limits on Main Steam Isolation Valve Leakage (TAC Nos. MA6405 and MA6406).

15. MicroShield, Version 5.0.3, Grove Engineering

16. NRC NUREG 1465, "Accident Source Terms for Light Water Reactors for Light-Water Nuclear Power Plants," Dated February 1995.

17. NRC NUREG/CR 5950, "Iodine Evolution and pH Control," Published December, 1992.

18. STARpH, "A Code for Evaluating Containment Water Pool pH during Accidents," R4, February 2000, Polestar Applied Technology, Inc.

19. NRC Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 134 to Facility Operating License N0. NPF-57, PSEG Nuclear, LLC, Atlantic City Electric Company, Hope Creek Generating Station, Docket No. 50-354, Dated October 3, 2001.

20. NRC Standard Review Plan 15.6.5, "Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Engineered Safety Feature Components Outside Containment," Dated July 1981.

21. NRC Standard Review Plan 6.5.2, "Containment Spray As a Fission Product Cleanup System," Revision 1, Dated July 1981.

22. AEB-98-03, "Assessment of Radiological Consequences For Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term," Dated December 9, 1998.

23. NRC Standard Review Plan 15.4.9, "Spectrum of Rod Drop Accidents (BWR)", Revision 2, Dated July 1981.

24. NRC Standard Review Plan 6.5.5, "Suppression Pool as a Fission Product Cleanup System," Dated December 1998.

25. NRC Standard Review Plan 6.4, "Control Room Habitability Systems," Dated July 1981.

ENCLOSURE 5

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

PROPOSED LICENSE AMENDMENT
ALTERNATIVE SOURCE TERM
UPDATED FINAL SAFETY ANALYSIS REPORT CHAPTER 14.6 MARKUPS

I. AFFECTED PAGE LIST

TVA anticipates that the following Chapter 14 pages will require revision by AST. The revised text has been highlighted with a line drawn through the deleted text and new or revised text indicated with a double underline.

A matrix identifying other sections in the UFSAR that are currently under evaluation for change is also provided in this enclosure. The final UFSAR changes will be completed as required by BFN procedures following approval of this change.

II. Marked Pages

See Attached

UFSAR Review Matrix

Chapter/ Appendix	Affected Sections*	Comments
1	1.2, 1.4, 1.6, 1.8	
2	None	
3	3.8	See Enclosure 3
4	4.5, 4.7, 4.8, 4.11	
5	5.2, 5.3	
6	6.1	
7	7.3, 7.12	
8	None	
9	None	
10	10.21	
11	None	
12	None	
13	None	
14	14.3, 14.6	Section 14.6 attached
A	None	
B	See marked-up TS pages	Technical Specifications
C	None	
D	Yes	
E	None	
F	None	
G	None	
H	N/A	Deleted
I	None	
J	None	
K	None	
L	None	
M	None	
N	None	

*These sections have been currently identified as requiring changes to support AST.

14.6.2.4 Fission Product Release From Fuel

The following assumptions were used in the initial calculation of fission product activity release from the fuel.

a. Eight hundred fifty fuel rods fail, per General Electric (GE) Licensing Topical Report, NEDO-31400A.

b. The reactor has been operating at design power (with a 1.02 uncertainty factor) ~~for 1400 days~~ with an average fuel burn-up of 35 to 37 GWd/MT prior to the accident. This assumption results in equilibrium concentration of fission products in the fuel. ~~Longer operating histories do not increase the concentration of longer lived fission products significantly.~~ The rods that have failed are assumed to have operated at a power peaking factor of 1.5[1].

c. Of the rods that fail, 0.~~00~~77% of the fuel melts, per NEDO-31400A. The following percentages of radioactive material are released to the reactor coolant from the failed fuel rods[8]:

~~Non-melted Rods~~	~~10% of Noble Gases~~	~~10% of Iodine~~
~~Melted Rods~~	~~100% of Noble Gases~~	~~50% of Iodine~~
Radionuclide Group	Non-Melted Rods	Melted Rods
Noble Gases	10%	100%
Iodine	10%	50%
Other Halogens	5%	30%
Alkali Metals	12%	25%
Tellerium Group	0%	5%
Barium, Strontium	0%	2%
Noble Metals	0%	0.25%
Cerium Group	0%	0.05%
Lanthanum Group	0%	0.02%

~~d. The fission product concentrations are the result of a nuclear analysis of the fuel assuming operation at design power for 1400 days times a factor of 1.02 for conservatism. per Reg. Guide 1.49.~~

8 Regulatory Guide 1.183 and NUREG-0800, Section 15.4.9.

e. ~~None of the solid fission products are released from the fuel. Because the fraction of solid fission product activity available for release from the fuel is negligible, this assumption is reasonable.~~

f. ~~The fission products produced during the nuclear excursion are neglected. The excursion is of such short duration that the fission products generated are negligible in comparison with the concentration of fission products already assumed present in the fuel.~~

~~Using the above assumptions, the following amounts of fission product activity are released from the failed fuel rods to the reactor coolant:~~

~~Noble gases (Xe,Kr)~~	~~2.423×10^6 Ci~~
~~Iodine~~	~~2.278×10^5 Ci~~

14.6.2.5 Fission Product Transport

The following assumptions were used in calculating the amounts of fission product activity transported from the reactor vessel to the main condenser ~~(initial core)~~:

a. ~~The recirculation flow rate is 25 percent of rated, and the steam flow to the condenser is five percent of rated. The 25 percent recirculation flow and five percent steam flow are the maximum flow rates expected when the reactor is being taken to power and the main condenser is still being evacuated by the mechanical vacuum pump (MVP). The recirculation flow rate is used in determining the volume of coolant in which the activity released from the fuel is deposited. The five percent steam flow rate is greater than that which would be in effect at the reactor power level assumed in the initial conditions for the accident. This assumption is conservative because it results in the transport of more fission products through the steam lines than would actually be expected.~~

a. ~~b.~~ Of the radioactive material released from the fuel, 100% of the noble gases ~~and~~ 10% of the iodines, and 1% of the remaining radionuclides, are assumed to reach the turbines and condensers[8].

c. ~~Water carryover in the main steam lines is assumed to be 0.1 percent of the total mass of steam transferred to the condenser. Measurements of the steam separation effected by the same types of separators used in this reactor vessel show that water carryover is less than 0.1 percent even at rated steam flow. The carryover fraction permits computation of the halogen activity carried to the main condenser in the water entrained in the steam.~~

b. ~~d~~ ~~None of the fission products released from the fuel is assumed to plate out.~~ Activity is assumed to be released from the core instantaneously to the condenser.

14.6.2.6 Fission Product Release to Environs

The following assumptions and initial conditions were used in the calculation of fission product activity released to the environs (initial core):

a. One reaching the condenser, 100% of the noble gases, 10% of the iodines, and 1% of the particulate radionuclides are available for release to the environment. Radioactive decay during holdup in the low pressure turbine and condenser is assumed.

b. ~~a.~~ The accident is assumed to occur while condenser vacuum is being maintained with the mechanical vacuum pump (MVP). During normal operation, vacuum is maintained with the steam-jet-air ejector, the discharge, from which, is through a holdup (time delay) and filter system. The assumed operation of the mechanical vacuum pump results in the discharge of the condenser activity directly to the environment via the elevated release point but without the benefits of holdup (decay) or filtration beyond the condenser.

c. ~~b.~~ All of the noble gas activity transferred to the condenser is assumed to be airborne in the condenser. The halogen and particulate activity transferred to the condenser experiences the removal effects of the condensate ~~and forms an equilibrium condition between the condensate and the vapor volume.~~ as described above.

d. ~~c.~~ The rate at which the condenser activity is discharged to the environment is dependent upon the free volume of the turbine and condenser and the discharge rate of the mechanical vacuum pump. The numerical values appropriate to these parameters are 187,000 ft^3 (low pressure turbine volume plus condenser free volume) and 1,850 cfm mechanical vacuum pump discharge rate.

e. ~~d.~~ A continuous ground level release of 10 cfm occurs at the base of the stack. The 10 cfm leakage mixes within the rooms at the base of the stack (34,560 ft^3, 50% of 69,120 ft^3 because of incomplete mixing).

f. ~~e.~~ Atmospheric dispersion coefficients, X/Q, for elevated releases under fumigation conditions, elevated releases under normal atmospheric conditions and ground level releases at the base of the stack are used. X/Q values applicable to the time periods, distances, and geometric relationships (offsite and control room) are shown in Table 14.6-8. Control room X/Q values for the base of the stack releases are calculated using the computer code ARCON96. For sites, such as BFN, with control room ventilation intakes that are close to the base of tall stacks, ARCON96 underpredicts the X/Q values for top of stack releases;

therefore, top of stack releases to the control room intakes are evaluated using the methods of Regulatory Guides 1.145 and 1.111.

g. ~~f.~~ The maximum control room X/Q for the top and bottom stack releases is used for each time period. The effective X/Q is a factor of two less than the values listed because of the dual air intake configuration of the control bay ventilation (i.e., one intake is not contaminated.)

Based upon these conditions, the fission product release rate to the environment is shown in Table 14.6-1.

14.6.2.7 Radiological Effects

The BFN analysis for the CRDA consists of two potential release paths; condenser leakage at 1% per day into the turbine building or through SJAE and offgas system as analyzed by the NEDO-31400A, and the MVP discharge as analyzed in accordance with Regulatory Guide 1.183. ~~SRP 15.4.9. The control room dose is divided by 2 because of the dilution effect of the dual air intake configuration of the control bay ventilation.~~ The "worst-case" radiological exposure resulting from the activity discharged from a CRDA and a Regulatory Guide 1.183 ~~an SRP 15.4.9~~ source term would be from the MVP release path. The resulting control room dose is less than the 10CFR50.67 limit of 5 Rem TEDE. ~~General Design Criteria (GDC) 19 limit of 5 Rem gamma, 30 Rem beta, and 30 Rem thyroid.~~ The ~~combined~~ EAB and LPZ doses from the MVP are well below the Regulatory Guide 1.183 ~~SRP 15.4.9~~ reference values of ~~75 REM thyroid and 6~~ 6.3 REM TEDE ~~whole body~~.

The dominant contributor to dose for the CRDA is Iodine 131 (I 131). Table 14.6-1 shows the I 131 activity in four locations (main condenser, stack room, control room, and environment) for the full 30 days of the dose calculation described above. This is an output of the RADTRAD computer code (NUREG/CR-6604) used for the CRDA dose analysis. Radioactive decay is considered in all locations except the environment (i.e., the environment represents a summation of all activity released). The environmental release totals approximately 10 percent of the activity initially reaching the main condenser. The main condenser is depleted of 95% of the activity by about 5 hours. This is consistent with an 1850 cfm exhaust rate and a 187,000 ft^3 volume (i.e. a release rate of about 0.6 volumes per hour).

14.6.3.3 Primary Containment Response

BFN Units 2 and 3 use the Mark I primary containment design. The main function of the Mark I containment design is to accommodate pressure and temperature conditions within the drywell resulting from a LOCA or a reactor blowdown through the MSRV discharge piping and, thereby, to limit the release of fission products to values which will ensure off-site dose rates below the ~~10 CFR 100~~ 10 CFR 50.67 limits. In the event of a pipe break in the drywell, water and/or steam from the reactor pressure vessel (RPV) are discharged into the drywell. The resulting increase in the drywell pressure forces the water and steam, along with non-condensable gases initially existing in the drywell, through the vents which connect the drywell to the suppression pool. During a reactor blowdown through the SRVs, the steam is directly discharged into the suppression pool. The reactor blowdown flow rate is dependent on the reactor initial thermal-hydraulic conditions, such as vessel dome pressure and the mass and energy of the fluid inventory in the RPV.

The long-term heatup of the suppression pool following a LOCA is governed by the capability of the Residual Heat Removal (RHR) System to remove decay heat which is transferred from the RPV to the suppression pool.

The Primary Containment System requirements are:

Design Pressure	56 psig
Design Temperature	281°F

Minimum containment overpressure following a LOCA and its affect on NPSH for Core Spray and RHR pumps is discussed in Chapter 6.5.5.

14.6.3.4 Fission Products Released to Primary Containment

The following assumptions and initial conditions were used in calculating the amounts of fission products released from the nuclear system to the drywell:

a. Source terms based on the ORIGEN computer code with a 1.02 multiplier per Regulatory Guide ~~1.49~~ 1.183.

b. The reactor has been operating at design power (~~3458~~ 3952 MWt) for a 24 month fuel cycle. The ~~total~~ average fuel burnup is ~~1,400 effective full power days (EFPDs).~~ 35 to 37 GWd/MT prior to the accident.

c. ~~One hundred percent of the equilibrium radioactive noble gas inventory developed as a result of such operation is released.~~ The radionuclides considered include those identified as being potentially important contributors to TEDE in NUREG/CR-6604.

d. ~~Twenty-five percent of the equilibrium radioactive iodine inventory developed as a result of such operation is released. Of this 25 percent, 91 percent is assumed to be elemental iodine, 5 percent in particulate form, and 4 percent in the form of organic iodides.~~ The core inventory release fractions, timing, and chemical form are those specified in Regulatory Guide 1.183. Table 14.6-7 gives the bounding core inventory of each isotope ~~in the primary containment available for leakage~~.

14.6.3.5 Fission Product Release From Primary Containment

Fission products are released from the primary containment to the secondary containment via primary containment penetration leakage at the Technical Specification leakage limit. Primary containment atmosphere is released via main steam isolation valve leakage to the high and low pressure turbines and the condenser. Primary containment atmosphere is released directly to the Standby Gas Treatment System during operation of the Containment Atmospheric Dilution (CAD) System. Primary containment atmosphere is released to the top of the stack via leakage of the hardened wetwell vent isolation valves. The Emergency Core Cooling Systems (ECCS) leak ~~reactor coolant~~ into the secondary containment. The following assumptions were used in calculating the amounts of fission products released from the primary containment:

a. The primary containment minimum free volume (drywell and wetwell) is ~~283,000~~ 278,400ft^3. The drywell volume is 159,000 ft^3 and the torus gas space volume is 119,400 ft^3. The drywell and torus gas space volumes are treated as separate volumes until after the activity release to the containment is complete and then these volumes are assumed to be well mixed. The activity release is entirely to the drywell.

b. The primary to secondary containment leak rate was taken as two percent volume per day (~~235~~ 232 cfh).

c. The four main steam lines are assumed to leak a total of ~~168~~ 150 scfh which ~~bounds~~ is the Technical Specification limit.

d. CAD System flow rate is 139 cfm for 24 hours at 10 days, 20 days, and 29 days.

e. The hardened wetwell vent isolation valves leak a total of 10 scfh to the top of the offgas stack. This leakage is assumed to begin at eight hours.

f. Five gpm ECCS leakage into secondary containment in accordance with NUREG-0800, Section 15.6.5, Appendix B.

g. No credit is taken for spray removal in the containment.

h. Natural removal rates for particulates in the drywell are based on the correlations of NUREG-CR-6604. For elemental iodine, the natural removal coefficients for removal of plateout are based on the expressions of SRP 6.5.2.

i. For the purposed of suppression pool pH control, the accident is assumed to be a recirculation line break.

Additionally, an analysis evaluated the suppression pool pH in the event of a DBA LOCA involving fuel damage. The objective of the analysis was to demonstrate that the suppression pool pH remains at or above 7.0; thus ensuring that the particulate iodine (Cesium Iodide - CsI) deposited into the suppression pool during this event does not re-evolve and become airborne as elemental iodine.

The calculation methodology was based on the approach outlined in NUREG-1465 and NUREG/CR-5950. Specifically, credit was taken for borate addition to the suppression pool water as a result of SLCS operation. The pH of the suppression pool water was then calculated using the STARpH code.

The initial effects on suppression pool pH come from rapid fission product transport and formation of cesium compound, which would result in increasing the suppression pool pH. As radiolytic production of nitric acid and hydrochloric acid proceeds and these acids are transported to the suppression pool over the first days of the event, the suppression pool water would become more acidic. The buffering effect of SLCS injection within several hours is sufficient to offset the effects of these acids that are transported to the pool. Sufficient sodium pentaborate solution is available to maintain the suppression pool pH at or above 7.0 for 30 days post accident.

14.6.3.6 Fission Product Release to Environs

Secondary Containment Releases

The fission product activity in the secondary containment at any time (t) is a function of the leakage rate from the primary containment, the volumetric discharge rate from the secondary containment and radioactive decay. During normal power operation, the secondary containment ventilation rate is 75 air changes per day; however, the normal ventilation system is turned off and the Standby Gas Treatment System (SGTS) is initiated as a result of low reactor water level, high drywell pressure, or high radiation in the Reactor Building. Any fission product removal effects in the secondary containment such as plateout are neglected. The fission product activity released to the environs is dependent upon the fission product inventory airborne in the secondary containment, the volumetric flow from the secondary containment, and the efficiency of the various components of the SGTS.

The following assumptions were used to calculate the fission product activity released to the environment from the secondary containment:

a. The primary containment atmosphere leakage to secondary containment mixes instantaneously and uniformly within the secondary containment.

b. The effective mixing volume of the secondary containment is 1,931,502 ft^3 (50% of total secondary containment volume).

c. The SGTS removes fission products from secondary containment. If only two of the SGTS trains are in operation (i.e., SGTS flow of 16,200 cfm), a short period exists at the start of the accident during which the secondary containment becomes pressurized relative to the outside environment. ~~During this short time period, a very small amount of secondary containment atmosphere (~35 ft³) will be released directly to the environment unfiltered from the Reactor Building.~~ However, negative pressure would be re-established in secondary containment prior to fission product release times specified by RG 1.183. Once the secondary containment pressure is reduced below atmospheric pressure, all releases from secondary containment to the environment are through the SGTS filters via the plant stack. If all three trains of SGTS are in operation (i.e., SGTS flow of 24,750 cfm), all releases to the environment from secondary containment are through the SGTS filters via the plant stack. The case with three trains in operation is the limiting condition.

d. The Containment Atmospheric Dilution (CAD) System operates for a period of 24 hours at a flow rate of 139 cfm at 10 days, 20 days, and 29 days post accident. This flow is filtered via the SGTS filters.

e. The ECCS systems leak reactor coolant directly to the secondary containment. The maximum water temperature is ~~177~~ less than 212°F. The ~~ECCS~~ volume available for mixing is ~~141,260~~ 1.41E5 ft^3. Ten percent of the iodine in the ECCS ~~water~~ leakage is assumed to become airborne.

f. Filter efficiency for the SGTS was taken as 90 percent for organic and 0% inorganic (elemental) iodine.

g. Release to the environment from the plant stack is composed of three flow paths. A continuous ground level release of 10 cfm occurs at the base of the stack. This flow results from SGTS leakage through the backdraft dampers in the base of the stack. Subsection 5.3.3, "Secondary Containment System

Description" describes the backdraft dampers. The 10 cfm leakage mixes uniformly within the room~~s~~ at the base of the stack (~~34,560~~ 50% of the room volume of 69,120 ft^3). The remaining SGTS flow exits the stack at a height of 183 meters above ground elevation. The hardened wetwell vent isolation valves leak a total of 10 scfh to the top of the offgas stack with a delay of 8 hours for the leakage to reach the stack. The hardened wetwell vent isolation valve leakage enters the stack above the divider deck and exits the top of the stack.

h. Fumigation conditions exist for ~~the first~~ 30 minutes when the post accident control room accumulated dose rate is maximum ~~post accident~~.

i. Atmospheric dispersion coefficients, X/Q, for elevated releases under fumigation conditions, elevated releases under normal atmospheric conditions and ground level releases at the base of the stack are used. X/Q values applicable to the time periods, distances, and geometric relationships (offsite and control room) are shown in Table 14.6-8. Control room X/Q values for the base of the stack releases are calculated using the computer code ARCON96. For sites, such as BFN, with control room ventilation intakes that are close to the base of tall stacks, ARCON96 underpredicts the X/Q values for top of stack releases; therefore, top of stack releases to the control room intakes are evaluated using the methods of Regulatory Guides 1.145 and 1.111.

j. The maximum control room X/Q for the top and bottom stack releases is used for each time period. Note that the effective X/Q is a factor of two less than the values listed because of the dual air intake configuration of the control bay ventilation. ~~The does from both the top and bottom of the stack released during 0 to 30 minutes are divided by a factor of 1.7 in accordance with the ICRP-30 conversion factors. The dose released during the 30 minute to 30 day time frame is similarly divided by 1.35.~~

Main Steam Isolation Valve Leakage Releases

~~The control room and offsite doses due to release of main steam isolation valve (MSIV) leakage were calculated in accordance with the BWROG methodology given in NEDC-31858-P-A, Revision 2~~

The leakage from primary containment via the MSIVs is transferred 1) to the main turbine (high pressure and low pressure) via the four steam lines and 2) to the condenser via the alternate leakage treatment (ALT) flow path formed by the steam line drain. The leakage from the turbine and condenser migrates to the turbine deck and subsequently is exhausted to the atmosphere via the turbine building roof vents with no credit for hold-up or removal in the turbine building. The path takes advantage of the large volume of the main steam lines and the condenser to hold up and plate out fission products in the MSIV leakage effluent. The following assumptions were used to calculate the fission product activity released to the environment from the turbine building:

a. The four main steam lines are assumed to leak a total of ~~168~~ 150 scfh which ~~bounds~~ is the Technical Specification limit. The direct leakage path to the turbines processes only 0.5% of the total leakage. The remainder goes to the condenser via the ALT flow path. The main steam piping from the outermost isolation valve up to the turbine stop valve, the bypass/drain piping to the main condenser and the main condenser will retain their structural integrity during and following a safe-shutdown earthquake (SSE).

b. ~~Iodine plateout and reentrainment are treated in accordance with the BWROG methodology.~~ Removal of particulate in the steam lines and the main condenser has been calculated using AEB 98-03 for (1) the well mixed efficiency model and (2) the deposition velocity. Removal in steam line control volumes upstream of the volume under consideration is accounted for by modifying the deposition velocity distribution. Elemental iodine removal in the steam lines is calculated using the Bixler model of NUREG/CR-6604. Elemental iodine removal in the main condenser is conservatively assumed to be the same as for particulate.

c. The free volume of the low pressure turbines is 51,000 ft^3 and the free volume of the condenser is taken as 90% of the total condenser volume of 136,000 ft^3.

d. No credit is taken for holdup in the turbine building.

e. Ground level atmospheric dispersion coefficients, X/Q, for releases from the turbine building roof vents applicable to the time periods, distances, and geometric relationships (offsite and control room) are shown in Table 14.6-8. Control room X/Q values are calculated using the computer code ARCON96.

~~f. ICRP-30 iodine dose conversion factors are applied.~~

14.6.3.7 Radiological Effects

The LOCA provides the most severe radiological releases to the primary and secondary containments and, thus, serves as the bounding design basis accident in determining post-accident offsite and control room personnel doses.

Offsite Doses

Offsite doses of interest resulting from the activity released to the environment as a consequence of the loss of coolant accident are the maximum 2-hour ~~whole body gamma dose, beta dose, and the thyroid inhalation~~ TEDE ~~dose at~~ for the exclusion area boundary (EAB) (1,465 meters), and the corresponding 30-day TEDE ~~doses~~ at the low population zone (LPZ) boundary (3,200 meters).

The offsite doses are calculated using ~~a combination of the STP and FENCEDOSE computer programs. The STP program models the fission product transport from the primary containment to release to the environment. The model accounts for fission~~

~~product decay, flow rates, filter absorption, dilution, release rates, and release points. The FENCEDOSE computer program models the atmospheric dispersion to the offsite receptor points~~ the RADTRAD code (NUREG/CR-6604) ~~by use of appropriate X/Qs and calculates the gamma, beta, and thyroid doses.~~ RADTRAD is a radiological consequence analysis code used to model plant control volumes for radionuclide transport and removal and account for atmospheric dispersion to offsite and control room locations by use of appropriate X/Qs.

The largest calculated total offsite dose is well within the 10 CFR ~~100~~ 50.67 limit ~~guideline values~~.

Control Room

The control room doses are calculated using ~~a combination of the STP and COROD computer programs~~ using RADTRAD (NUREG/CR-6604). ~~The STP program models the fission product transport from the primary containment release to the environment. The model accounts for fission product decay, flow rates, SGTS filter absorption, dilution, release rates, and release points.~~ The ~~COROD computer program~~ model accounts for the atmospheric dispersion to the dual control room intakes by use of appropriate X/Qs and models the control bay habitability zone filtered pressurization flow (3000 cfm), unfiltered inleakage (3717 cfm), Control Room Emergency Ventilation System (CREVS) filter absorption (90% particulate ~~organic and inorganic iodine~~), occupancy times, breathing rates in accordance with Regulatory Guide 1.~~3~~ 183 and calculates the ~~gamma, beta, and thyroid~~ TEDE ~~doses~~. Atmospheric dispersion coefficients are based on release point, geometric relationship of the release point, and receptor and atmospheric conditions based on site specific meteorological data. The model accounts ~~The COROD computer code calculates the gamma dose by a typical point kernel methodology accounting~~ for the control room geometry (210,000 ft^3).

The direct gamma-dose contribution from the piping inside secondary containment~~,~~ and the secondary containment atmosphere~~, and the cloud dose~~ are included. One section of core spray piping in each unit is routed just outside the common Control Building/Reactor Building wall. This piping will be carrying ~~pressure~~ suppression ~~chamber~~ pool water in the event of a LOCA.

All of these exposure mechanisms (filtered pressurization flow, unfiltered inleakage, ~~cloud dose~~, and direct dose) are combined to produce a total control room dose for the duration of the accident. ~~It was determined that the differences between the case with two SGTS fans in operation with a small amount of unfiltered secondary containment release and the case with three SGTS fans in operation with all releases being filtered and via the plant stack are negligible.~~ Since CREVS has dual air intakes placed on opposite sides of the control building and can function with a single active failure in the inlet isolation system, in accordance with NUREG-0800, the control room dose is divided by a factor of 2 to account for dilution effects. The 30 day integrated post-accident doses in the control room are within the limits of 5 REM TEDE ~~whole body gamma dose, 30 REM beta, and 30 REM to the thyroid~~ as specified in 10 CFR 50.67~~, Appendix A General Design Criteria 19~~.

14.6.4 Refueling Accident

The current safety evaluation for the Refueling Accident is contained in the licensing topical report for nuclear fuel, "General Electric Standard Application For Reactor Fuel," NEDE-24011-P-A, and subsequent revisions thereto. Accidents that result in the release of radioactive materials directly to the secondary containment are events that can occur when the primary containment is open. A survey of the various plant conditions that could exist when the primary containment is open reveals that the greatest potential for the release of radioactive material exists when the primary containment head and reactor vessel head have been removed. With the primary containment open and the reactor vessel head off, radioactive material released as a result of fuel failure is available for transport directly to the reactor building.

Various mechanisms for fuel failure under this condition have been investigated. Refueling Interlocks will prevent any condition which could lead to inadvertent criticality due to control rod withdrawal error during refueling operations when the mode switch is in the Refuel position. The Reactor Protection System is capable of initiating a reactor scram in time to prevent fuel damage for errors or malfunctions occurring during deliberate criticality tests with the reactor vessel head off. The possibility of mechanically damaging the fuel has been investigated.

The design basis accident for this case is one in which one fuel assembly is assumed to fall onto the top of the reactor core.

The discussion in Subsections 14.6.4.1 and 14.6.4.2 provides the analyses for the dropping of a 7 x 7 assembly and a 8 x 8 assembly. The analyses for all current General Electric product line fuel bundle designs are contained in supplements to NEDE-24011-P-A. The NEDE evaluates each new fuel design against the 7x7 fuel design for the original core load. The 7x7 fuel handling accident resulted in 111 failed fuel rods. ~~For the 8x8 fuel design, the activity released due to a fuel handling accident will be less than 88% of the activity released by the original 7x7 fuel design. For the 9x9 fuel design the activity will be less than 83.5% of the activity released by the original 7x7 fuel design.~~ <u>Evaluation of other fuel types has been performed as a comparison of the fuel damage to the 7x7 fuel design. Fuel types evaluated include 8x8, 8x8R, 9x9, GE-14 (10x10) and Framatome Atrium 10 (A-10). The activity release for each of these fuel types is bounded by the 7x7 case.</u> The historical and current calculated doses are much less than the regulatory guidelines.

14.6.4.3 Fission Product Release From Fuel

The radiological dose consequences resulting from a refueling accident have been evaluated using Alternative Source Terms (AST) in accordance with 10 CFR 50.67 and the guidelines of Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

~~The following radiological consequences are based on 8x8 fuel.~~ Fission product release estimates for the accident are based on the following assumptions:

a. ~~The reactor fuel has an average irradiation time of 1000 days at design power up to 24 hours prior to the accident. This assumption results in an equilibrium fission product concentration at the time the reactor is shut down. Longer operating histories do not significantly increase the concentration of the fission products of concern.~~ The reactor has been operating at design power (3952 MWt) for a 24 month fuel cycle. The average fuel burnup is 35 to 37 GWd/MT prior to the accident. A decay time of 24 hours is assumed. The 24-hour decay time allows time for the reactor to be shut down, the nuclear system depressurized, the reactor vessel head removed, and the reactor vessel upper internals removed. It is not expected that these evolutions could be accomplished in less than 24 hours.

b. The activity in the fuel bundle is determined using the ORIGEN code at a core power of 4031 MWt modified with a power peaking factor of 1.5 and Regulatory Guide ~~1.49~~ 1.183 power factor of 1.02 with a decay of 24 hours.

~~c. Due to the negligible particulate activity available for release in the fuel plenums or from the unmelted fuel, none of the solid fission products are assumed to be released from the fuel.~~

~~d~~c. One hundred eleven fuel rods are assumed to fail. This was the conclusion of the analysis of mechanical damage to the fuel based on the GE 7x7 fuel design.

14.6.4.4 Fission Product Release to Secondary Containment

The following assumptions were used to calculate the fission product release to the secondary containment (per Regulatory Guide 1.183):

a. Fraction of Fuel Rod Inventory Released (infinite decontamination for nuclides other than iodine and noble gases):

Noble Gases (Except Kr 85)	~~10~~ 5 percent
Kr 85	~~30~~ 10 percent
Iodines (except I-131)	~~10~~ 5 percent
I-131	8 percent

b. Iodine Decontamination Factor in Reactor Cavity Pool Water — ~~100~~ 500 elemental, 1 organic

c. Iodine Species — 99.85% elemental, 0.15% organic

14.6.4.5 Fission Product Release to Environs

The following assumptions and initial conditions are used in calculating the dose existing at the exclusion area boundary ~~and~~ , at the low population zone, and to the control room operators due to fission product release.

~~a. High radiation levels in the reactor building will isolate the normal ventilation system and actuate the Standby Gas Treatment System. The isolation dampers were assumed to close in 10 seconds.~~

~~b. Since the refueling accident does not result in the release of any liquid or vapor to the secondary containment, the normal environmental condition existing prior to the accident will also exist after the accident except for the addition of the released fission products. The relative humidity in the secondary containment will, therefore, be considerably below any levels which may be detrimental to the Standby Gas Treatment System. However, air flowing through the filter system, is heated approximately 14°F above the mixture entering the system, reducing the relative humidity from 100 percent to 70 percent or less.~~

~~c. Standby Gas Treatment System Filter Efficiency — 0.90~~

~~d. Height of the Main Stack — 183 meters~~

~~e. Distance to Exclusion Area Boundary — 1,465 meters~~

~~f. Distance to Low Population Zone — 3,200 meters~~

~~g. Mixing Air Volume — 4,900 FT^3~~

~~h. Ventilation Air Flow Prior to Damper Isolation — 20,000 CFM~~

a. The release is assumed to be an instantaneous ground level release to the environment with no holdup time in secondary containment. Accordingly, no credit is taken for filtering by the standby gas treatment system and no credit is taken for an elevated release at the main stack.

b. No credit is taken for isolation of the control room nor for any filtering by the control room emergency ventilation system.

c. The X/Q for the control room is reduced by 50% to reflect the credit for dual control room air intakes as allowed by Standard Review Plan Section 6.4.

d. Control Room Free Volume 210,000 ft^3

The design basis fuel handling accident assumes that during the refueling period a fuel bundle is dropped into the reactor cavity pool. The dropped fuel bundle strikes additional bundles in the reactor core fracturing 111 fuel pins (assuming GE 7x7 fuel design). ~~Ten percent of the halogen isotopes inventory plus 10 percent of all noble gases inventory (except Kr-85 which is 30 percent of this inventory)~~ The inventory described above will be released from the fractured fuel rods. ~~An overall effective~~ ~~d~~Decontamination factors of ~~100 is~~ 500 for elemental iodine and 1 for organic iodine are applicable for iodine released at depth under water. The radioactive releases to the air space above the pool are released ~~through the refueling zone ventilation and the Standby Gas Treatment Systems~~ instantaneously to the atmosphere with no holdup in secondary containment and no filtering by the standby gas treatment system. The assumptions used to evaluate the fuel handling design basis accident event are defined in Nuclear Regulatory Commissions Regulatory Guide ~~1.25~~ 1.183. Further guidance is contained in the standard review plans in NUREG-800, Section ~~15.7.4~~ 15.0.1.

~~In order to evaluate the effect of refueling zone ventilation damper closure time, the analysis includes doses from air bypassing the Standby Gas Treatment System. The bypass is occurring through the Refueling Zone Ventilation System. For this evaluation, it is assumed that the portion of the ventilation system dedicated to the reactor vessel pool and the spent fuel storage pool provides the bypass flow. The gases released from the damaged fuel bundles are assumed to be confined to an air volume bounded by the perimeter of the pool and mixed to a height of no more than 4 feet above the pool. The activity released to the environment before the dampers close is taken from the air volume over the pool expelled through the ventilation system.~~ The total activity released is greater for a fuel handling accident in the reactor cavity pool than for an accident in the fuel storage pool. Normally, the number of fuel rods fractured in a drop into the reactor vessel pool is slightly larger than the number of rods fractured in a drop into the storage pool. This provides a bigger source for the vessel event. ~~However, the ventilation flow from the storage pool area is twice the size of the flow from the reactor vessel area. The difference in flows transports more activity to the environment in a given time period. Therefore, for conservatism the number of rods damaged and resulting activity released is based on a fuel handling accident in the reactor cavity, and the mixing volume and ventilation is based on a release over the spent fuel pool.~~

~~The bypass flow not only bypasses the SGTS filters, it is also released from a roof vent rather than the main stack. The atmospheric dispersion, X/Q, of releases from the top of the stack is significantly smaller than the atmospheric dispersion factors for the roof vent releases. The result of this change is to make the dose contribution from the roof~~

~~vent releases more important than if all releases were through the stack. Almost all the dose is from the roof vent release.~~

The fuel handling accident was evaluated using ~~the STP, FENCEDOSE, and COROD computer programs~~ RADTRAD as described in Section 14.6.3.7. The X/Q values based on the refueling vents from 0 - 2 hours were used in computing the dose consequences of this release. ~~The calculations simulate an initial time period without filtration of the releases. Following the initial time period, the releases are filtered. Computations were prepared with an atmospheric dispersion, X/Q, for elevated releases and with X/Q data for ground level releases appropriate for the EAB and LPZ boundaries. Control room X/Q values for the base of the stack releases are calculated using the computer code ARCON96. For sites, such as BFN, with control room ventilation intakes that are close to the base of tall stacks, ARCON96 underpredicts the X/Q values for top of stack releases; therefore, top of stack releases to the control room intakes are evaluated using the methods of Regulatory Guides 1.145 and 1.111.~~

~~The final dose evaluations become the dose contributions from the initial ground level release plus the contribution from the release of the balance of the activity through the stack (base and top).~~

14.6.4.6 Radiological Effects

The radiological exposures following the refueling accident have been evaluated in the control room, at the site boundary, and at the LPZ boundary. The calculated dose assumes that all of the ~~bypass~~ activity is exhausted instantaneously through a roof vent; with no credit for holdup time nor filtering by the SGTS ~~and after the dampers close, the activity is processed through the SGTS and the plant stack. The control room dose is divided by 2 because of the dilution effect of the dual air intake configuration of the control bay ventilation.~~

Boundary dose resulting from design basis accident events has been judged by comparing the dose to the ~~dose in 10 CFR 100, Reactor Site Criteria~~ 10 CFR 50.67, "Accident Source Term," limits. This regulation uses radiation doses of ~~300 rem to the thyroid through inhalation, 300 rem whole body beta, and 25 rem whole body gamma as guides~~ 25 rem TEDE for doses to the public and 5 rem TEDE for the control room as guides under accident conditions. ~~Fuel handling accidents in the past have been judged as having acceptable consequences if the dose is a small part of 10 CFR 100 limits.~~ In the standard review plan, NUREG-800, ~~a small part has been defined as~~ the limits for doses to the public are reduced by 25 percent to 6.3 rem TEDE. The calculated doses are much less than the guidelines (~~≤ 6.25 rem gamma, ≤ 75 rem beta and inhalation~~ ≤ 6.3 rem TEDE for EAB & LPZ and ≤ 5 rem TEDE for the control room).

~~As part of the BFN Power Uprate licensing analyses, the impact of Power Uprate on the radiological consequences from a Refueling Accident was also evaluated. It is concluded that the radiological consequences of this postulated accident will continue to meet regulatory requirements at a rated thermal power of 3458 megawatts and, therefore, are acceptable.~~

14.6.5.2 Radioactive Material Release

14.6.5.2.1 Assumptions

The following assumptions are used in the calculation of the quantity and types of radioactive material released from the nuclear system process barrier outside the secondary containment:

a. The amounts of steam and liquid discharged are as calculated from the analysis of the nuclear system transient.

b. The concentrations of biologically significant radionuclides contained in the coolant discharged as liquid (which subsequently flashes to steam) and the coolant discharged as steam are based on the ANSI/ANS-18.1-1984, "Radioactive Source Term for Normal Operation of Light Water Reactors" methodology. The halogens considered are I-131, I-132, I-133, I-134, and I-135. The values obtained by the ANSI/ANS-18.1 evaluation are then scaled to represent a dose equivalent I-131 concentration of 32 μCi/~~cc~~gm which is greater than the 26 μCi/gm maximum Technical Specification limit and 10 times the equilibrium value for continued full power operation allowed by Technical Specifications. ~~Since this value is 10 times the equilibrium value for continued full power operation allowed by Technical Specifications and several orders of magnitude higher than normal reactor coolant concentrations, considerable conservatism is included in the analysis.~~

c. The concentration of noble gases leaving the reactor vessel at the time of the accident are based on the ANSI/ANS-18.1 concentrations with an appropriate scaling based on NEDO-10871, "Technical Derivation of BWR 1971 Design Basis Radioactive Material Source Terms".

d. It is assumed that the main steam isolation valves are fully closed at 5.5 seconds after the pipe break occurs. This allows 500 milliseconds for the generation of the automatic isolation signal and 5 seconds for the valves to close. The valves and valve control circuitry are designed to provide main steam line isolation in no more than 5.5 seconds. The actual closure time setting for the isolation valves is less than 5 seconds.

e. Due to the short half-life of nitrogen-16 the radiological effects from this isotope are of no major concern and are not considered in the analysis.

f. Atmospheric dispersion coefficients, X/Q, for elevated releases under fumigation conditions, elevated releases under normal atmospheric conditions and ground level releases at the base of the stack are used. An instantaneous ground level puff release X/Q value for the control room was accomplished in accordance with Regulatory Guide 1.183, Appendix D. X/Q values applicable to the time periods, distances and geometric relationships (offsite and control room) are shown in Table 14.6-8. ~~Control room X/Q values are calculated using the~~

~~computer code ARCON96.~~

14.6.5.2.2 Fission Product Release From Break

Using the above assumptions, the following amounts of radioactive materials are released from the nuclear system process barrier:

Noble gases	~~1.413~~ 1.342 x 10^3 Ci
Iodine 131	~~5.287~~ 5.254 x 10^1 Ci
Iodine 132	~~4.676~~ 4.737 x 10^2 Ci
Iodine 133	~~3.546~~ 3.533 x 10^2 Ci
Iodine 134	~~8.328~~ 8.549 x 10^2 Ci
Iodine 135	~~5.021~~ 5.031 x 10^2 Ci

The above releases take into account the total amount of liquid released as well as the liquid converted to steam during the accident.

14.6.5.2.3 Instantaneous Ground Level Puff Release

This X/Q reflects an instantaneous ground level puff release to the atmosphere. This is the instantaneous puff release of a coherent steam mass in which the only dispersion initially available is that due to the expansion of the steam from the blowdown itself.

The puff release calculation uses plant parameters for the MSLBA (e.g., mass of liquid-steam mixture released, timing of release, temperature of the liquid-steam mixture) to obtain the initial conditions of the released steam puff. The steam puff is treated as a 'bubble' with a given transit time up to and across the control room intake. Mixing of the steam with surrounding air reduces the bubble's buoyancy but also increases dilution. Different bubble shapes and degrees of air entrainment are considered, and the worst case is used (i.e., minimum dilution).

The bubble is assumed to be released from the turbine building at a distance from the nearest control room intake that is exceeded by 90% of the potential release locations. No credit is taken for wind direction; i.e., it is assumed that the centerline of the bubble trajectory always passes over one control room intake.

~~14.6.5.2.3 Steam Cloud Movement~~

~~The following initial conditions and assumptions are used in calculating the movement of the steam cloud:~~

~~a. Additional flashing to steam of the liquid exiting from the steam line break will occur due to its superheated condition in the atmosphere.~~

~~b. The pressure buildup inside the turbine building will cause the blowout panels to function, resulting in release of the steam cloud in a matter of seconds.~~

~~c. Steam cloud rise as predicted by the following equation could vary between 100 and 600 meters depending upon the assumptions made regarding wind speed.[3]~~

$$\cancel{h = \frac{11Q^{1/3}}{u}}$$

~~where:~~

~~h = Height of cloud rise (ft)~~
~~u = Wind speed (ft/sec)~~
~~Q = Heat output of cloud (cal/sec)~~

~~While the effect of cloud rise is a physical reality, this effect has been neglected for this accident and the assumption is made that the steam cloud does not attain an elevation greater than the height of the turbine building.~~

~~All of the activity released from the reactor vessel to the turbine building is conservatively assumed to escape to the environment.~~

14.6.5.3 Radiological Effects

The control room dose is divided by 2 because of the dilution effect of the dual air intake configuration of the control bay ventilation. Shine due to radioisotopes in the Turbine Building are taken into account by adding to the gamma and TEDE in the control room dose. The shine is not divided by 2. The control room operator doses due to a MSLB are less than the 10CFR50.67 limit of 5 rem TEDE. The offsite doses are less than the 10CFR50.67 limit of 25 rem TEDE for the maximum Technical Specification reactor coolant (32 μCi/gm I-131 equivalent). Also, the offsite doses are less than 10% of the 10CFR50.67 limits (2.5 rem TEDE) for the maximum equilibrium reactor coolant (3.2 μCi/gm). ~~These values are well below (< 10% of) the guideline doses of 25 rem gamma and 300 rem beta and thyroid set forth in 10 CFR 100.~~

~~Since all of the activity is released to the environment in the form of a puff, the doses indicated are maximum values regardless of what dose period is being evaluated.~~

It is concluded that no danger to the health and safety of the public results as a consequence of this accident.

~~[3] Singer, I. A., Frizzola, J. A., Smith M. E., "The Prediction of the Rise Of A Hot Cloud From Field Experiments, "Journal of the Air Pollution Control Association, November, 1964.~~

~~Table 14.6-1~~

~~CONTROL ROD DROP ACCIDENT~~

~~FISSION PRODUCT RELEASE TO ENVIRONMENT~~

~~(CURIES)~~

~~Isotope~~	~~30 Minutes*~~	~~2 Hours~~	~~8 Hours~~	~~24 Hours~~	~~4 Days~~	~~30 Days~~
~~I-131~~	~~6.773E2~~	~~1.152E3~~	~~7.740E2~~	~~2.504E1~~	~~5.772E0~~	~~1.640E0~~
~~I-132~~	~~9.120E2~~	~~1.191E3~~	~~4.175E2~~	~~2.006E0~~	~~5.167E-4~~	~~4.604E-14~~
~~I-133~~	~~1.361E3~~	~~2.253E3~~	~~1.412E3~~	~~3.735E1~~	~~2.957E0~~	~~7.893E-2~~
~~I-134~~	~~1.256E3~~	~~1.000E3~~	~~1.577E2~~	~~4.005E-2~~	~~2.674E-9~~	~~0~~
~~I-135~~	~~1.254E3~~	~~1.947E3~~	~~1.037E3~~	~~1.720E1~~	~~2.119E-1~~	~~3.191E-5~~
~~Kr-85m~~	~~1.700E4~~	~~2.527E4~~	~~1.208E4~~	~~1.459E2~~	~~6.245E-1~~	~~2.595E-6~~
~~Kr-85~~	~~1.019E3~~	~~1.739E3~~	~~1.178E3~~	~~3.906E1~~	~~1.045E1~~	~~4.192E0~~
~~Kr-87~~	~~2.961E4~~	~~3.161E4~~	~~6.994E3~~	~~7.797E0~~	~~3.114E-5~~	~~8.036E-23~~
~~Kr-88~~	~~4.459E4~~	~~6.115E4~~	~~2.403E4~~	~~1.633E2~~	~~1.172E-1~~	~~6.089E-10~~
~~Kr-89~~	~~9.692E3~~	~~1.043E1~~	~~1.294E-8~~	~~0~~	~~0~~	~~0~~
~~Xe-131m~~	~~7.870E2~~	~~1.340E3~~	~~9.068E2~~	~~4.090E1~~	~~5.134E0~~	~~1.367E2~~
~~Xe-133m~~	~~4.454E3~~	~~7.521E3~~	~~5.003E3~~	~~2.836E2~~	~~1.873E1~~	~~2.057E1~~
~~Xe-133~~	~~1.383E5~~	~~2.351E5~~	~~1.585E5~~	~~7.592E3~~	~~4.405E3~~	~~5.909E2~~
~~Xe-135m~~	~~1.707E4~~	~~8.900E3~~	~~1.366E4~~	~~1.249E4~~	~~2.867E3~~	~~1.279E0~~
~~Xe-135~~	~~4.924E4~~	~~8.111E4~~	~~5.638E4~~	~~1.262E4~~	~~2.723E3~~	~~1.446E0~~
~~Xe-137~~	~~2.497E4~~	~~8.166E1~~	~~2.829E-6~~	~~0~~	~~0~~	~~0~~
~~Xe-138~~	~~6.318E4~~	~~1.299E4~~	~~6.568E1~~	~~4.276E-8~~	~~0~~	~~0~~

~~* = Fumigation~~

~~m = metastable state~~

Table 14.6-1

Iodine-131 Activity (Ci) by Location as Function of Time for CRDA

Time - hrs	Main Cond	Stack Rm	Control Rm	Environment
0	2.99E+04	0.00E+00	0.00E+00	0.00E+00
0.4	2.35E+04	3.39E+00	2.32E-02	6.27E+02
0.5	2.22E+04	4.11E+00	2.57E-02	7.60E+02
0.8	1.85E+04	6.03E+00	1.45E-02	1.12E+03
1.1	1.55E+04	7.62E+00	8.20E-03	1.42E+03
1.4	1.29E+04	8.93E+00	4.65E-03	1.67E+03
1.7	1.08E+04	1.00E+01	2.65E-03	1.88E+03
2	9.05E+03	1.09E+01	1.52E-03	2.06E+03
2.3	7.56E+03	1.16E+01	8.60E-04	2.20E+03
2.6	6.32E+03	1.22E+01	4.89E-04	2.33E+03
2.9	5.29E+03	1.27E+01	2.80E-04	2.43E+03
3.2	4.42E+03	1.31E+01	1.62E-04	2.51E+03
3.5	3.69E+03	1.34E+01	9.49E-05	2.59E+03
3.8	3.09E+03	1.36E+01	5.65E-05	2.65E+03
4.1	2.58E+03	1.38E+01	3.44E-05	2.70E+03
4.4	2.16E+03	1.40E+01	2.16E-05	2.74E+03
4.7	1.80E+03	1.41E+01	1.40E-05	2.77E+03
5	1.51E+03	1.41E+01	9.44E-06	2.80E+03
5.3	1.26E+03	1.42E+01	6.62E-06	2.83E+03
5.6	1.05E+03	1.42E+01	4.82E-06	2.85E+03
5.9	8.81E+02	1.42E+01	3.62E-06	2.86E+03
6.2	7.37E+02	1.42E+01	2.80E-06	2.88E+03
6.5	6.16E+02	1.42E+01	2.22E-06	2.89E+03
6.8	5.15E+02	1.41E+01	1.78E-06	2.90E+03
7.1	4.30E+02	1.41E+01	1.45E-06	2.91E+03
7.4	3.60E+02	1.40E+01	1.19E-06	2.92E+03
7.7	3.01E+02	1.40E+01	9.81E-07	2.92E+03
8	2.51E+02	1.39E+01	8.13E-07	2.93E+03
8.3	2.10E+02	1.39E+01	5.80E-07	2.93E+03
8.6	1.76E+02	1.38E+01	4.29E-07	2.93E+03
8.9	1.47E+02	1.37E+01	3.28E-07	2.94E+03
9.2	1.23E+02	1.36E+01	2.56E-07	2.94E+03
9.5	1.03E+02	1.36E+01	2.04E-07	2.94E+03
9.8	8.58E+01	1.35E+01	1.65E-07	2.94E+03
10.1	7.17E+01	1.34E+01	1.35E-07	2.94E+03
10.4	6.00E+01	1.33E+01	1.11E-07	2.94E+03
24	1.78E-02	1.01E+01	0	2.95E+03
96	0	2.22E+00	0	2.95E+03
720	0	0	0	2.95E+03

~~Table 14.6-7~~

~~INVENTORY IN PRIMARY CONTAINMENT AVAILABLE FOR LEAKAGE~~

~~Isotope~~	~~Activity (Ci)~~	~~λ (hr^{-1})~~	~~Isotope~~	~~Activity (Ci)~~	~~λ (hr^{-1})~~
~~^{131}I~~	~~9.0029E7~~	~~3.59E-3~~	~~^{85}Kr~~	~~1.350E6~~	~~7.34E-6~~
~~^{132}I~~	~~1.3008E8~~	~~3.01E-1~~	~~^{87}Kr~~	~~4.4810E7~~	~~5.47E-1~~
~~^{133}I~~	~~1.8221E8~~	~~3.30E-2~~	~~^{88}Kr~~	~~6.3030E7~~	~~2.48E-1~~
~~^{134}I~~	~~1.9978E8~~	~~7.88E-1~~	~~^{89}Kr~~	~~7.6530E7~~	~~1.31E1~~
~~^{135}I~~	~~1.7069E8~~	~~1.03E-1~~	~~$^{131}Xe^m$~~	~~1.0500E6~~	~~2.41E-3~~
~~$^{131}I^*$~~	~~3.7512E6~~	~~3.59E-3~~	~~$^{133}Xe^m$~~	~~5.9560E6~~	~~1.28E-2~~
~~$^{132}I^*$~~	~~5.4200E6~~	~~3.01E-1~~	~~^{133}Xe~~	~~1.8470E8~~	~~5.48E-3~~
~~$^{133}I^*$~~	~~7.5920E6~~	~~3.30E-2~~	~~$^{135}Xe^m$~~	~~3.7610E7~~	~~2.65E0~~
~~$^{134}I^*$~~	~~8.3240E6~~	~~7.88E-1~~	~~^{135}Xe~~	~~6.6100E7~~	~~7.57E-2~~
~~$^{135}I^*$~~	~~7.1120E6~~	~~1.03E-1~~	~~^{137}Xe~~	~~1.6550E8~~	~~1.09E1~~
~~$^{85}Kr^m$~~	~~2.3510E7~~	~~1.58E-1~~	~~^{138}Xe~~	~~1.5520E8~~	~~2.93E0~~

~~Note: * denotes organic form, m denotes metastable state~~

Table 14.6-7

BOUNDING CORE INVENTORY

Isotope	Ci/MWt t = 0	Ci/MWt t = 24 hr
CO58	1.430E+02	1.416E+02
CO60	1.425E+02	1.424E+02
KR83M	3.432E+03	1.387E+01
KR85	3.601E+02	3.601E+02
KR85M	7.329E+03	1.811E+02
RB86	6.372E+01	6.141E+01
KR87	1.446E+04	3.051E-02
KR88	2.009E+04	5.743E+01
KR89	2.521E+04	0.000E+00
SR89	2.786E+04	2.748E+04
SR90	3.165E+03	3.165E+03
Y90	3.283E+03	3.273E+03
SR91	3.487E+04	6.103E+03
Y91	3.583E+04	3.564E+04
SR92	3.677E+04	7.922E+01
Y92	3.696E+04	1.168E+03
Y93	4.147E+04	8.084E+03
ZR95	4.880E+04	4.822E+04
NB95	4.897E+04	4.897E+04
ZR97	4.953E+04	1.851E+04
MO99	5.088E+04	3.956E+04
TC99M	4.454E+04	3.772E+04
RU103	4.094E+04	4.018E+04
RU105	2.710E+04	6.615E+02
RH105	2.559E+04	1.840E+04
RU106	1.488E+04	1.486E+04
SB127	2.796E+03	2.369E+03
TE127	2.773E+03	2.580E+03
TE127M	3.721E+02	3.719E+02
SB129	8.457E+03	1.952E+02
TE129	8.326E+03	1.236E+03
TE129M	1.615E+03	1.590E+03
TE131M	5.155E+03	2.976E+03
I131	2.669E+04	2.481E+04

Isotopes	Ci/MWt t = 0	Ci/MWt t = 24 hr
XE131M	3.544E+02	3.487E+02
TE132	3.829E+04	3.089E+04
I132	3.885E+04	3.184E+04
I133	5.534E+04	2.559E+04
XE133	5.504E+04	5.303E+04
XE133M	1.734E+03	1.562E+03
I134	6.141E+04	1.450E-03
CS134	5.703E+03	5.697E+03
I135	5.250E+04	4.189E+03
XE135	1.971E+04	1.429E+04
XE135M	1.135E+04	6.823E+02
CS136	1.941E+03	1.841E+03
XE137	5.023E+04	0.000E+00
CS137	4.037E+03	4.037E+03
BA137M	3.829E+03	3.810E+03
XE138	4.757E+04	1.172E-26
BA139	4.930E+04	4.170E-01
BA140	4.909E+04	4.644E+04
LA140	5.231E+04	5.079E+04
LA141	4.498E+04	7.085E+02
CE141	4.535E+04	4.463E+04
LA142	4.397E+04	1.035E+00
CE143	4.245E+04	2.597E+04
PR143	4.113E+04	4.075E+04
CE144	3.810E+04	3.810E+04
ND147	1.806E+04	1.698E+04
NP239	5.201E+05	3.902E+05
PU238	2.805E+02	2.805E+02
PU239	1.234E+01	1.238E+01
PU240	1.730E+01	1.730E+01
PU241	4.450E+03	4.448E+03
AM241	5.449E+00	5.470E+00
CM242	1.234E+03	1.234E+03
CM244	5.697E+01	5.697E+01

Table 14.6-8

(Sheet 1)

VALUES FOR X/Q FOR ACCIDENT DOSE CALCULATIONS

Time Period	Control Room (sec/m^3)		Site Boundary (sec/m^3)	LPZ Boundary (sec/m^3)
Top of Stack Releases (LOCA,& CRDA, ~~& FHA~~)	U1 Intake	Unit 3 Intake		
0-0.5 hrs*	3.40E-5	3.02E-5	2.35E-5	1.26E-5
0.5-2 hrs	9.08E-13	1.41E-7	1.19E-6	1.13E-6
2-8 hrs	3.41E-13	4.50E-8		5.75E-7
8-24 hrs	2.09E-13	2.54E-8		4.10E-7
1-4 days	7.21E-14	7.36E-9		1.97E-7
4-30 days	1.57E-14	1.24E-9		6.88E-8
Base of Stack Releases (LOCA, & CRDA, ~~& FHA~~)				
0-2 hrs	2.00E-4	8.60E-5	2.62E-4	1.31E-4
2-8 hrs	1.28E-4	6.46E-5		6.61E-5
8-24 hrs	5.72E-5	2.80E-5		4.69E-5
1-4 days	4.05E-5	2.00E-5		2.23E-5
4-30 days	3.09E-5	1.53E-5		7.96E-6

Table 14.6-8

(Sheet 2)

VALUES FOR X/Q FOR ACCIDENT DOSE CALCULATIONS

Time Period	Control Room (sec/m^3)		Site Boundary (sec/m^3)	LPZ Boundary (sec/m^3)
Refueling Vent Releases (FHA Only)	U1 Intake	Unit 3 Intake		
0-2 hrs	4.60E-4	**		
Turbine Building Exhaust Release (MSLB Only)				
0-2 hrs	~~3.22E-4~~	~~**~~	2.62E-4	1.31E-4
~~2-8 hrs~~	~~2.77E-4~~	~~**~~		~~6.61E-5~~
~~8-24 hrs~~	~~1.31E-4~~	~~**~~		~~4.69E-5~~
~~1-4 days~~	~~7.91E-5~~	~~**~~		~~2.23E-5~~
~~4-30 days~~	~~6.10E-5~~	~~**~~		~~7.96E-6~~
~~46 Seconds~~	~~4.60E-4~~	~~**~~		

**Bounded by the Unit 1 Intake

Time Period	U1 Intake	Unit 3 Intake	Site Boundary	LPZ Boundary
Turbine Building Roof Ventilators Release (Post LOCA MSIV Leakage Only)				
0-2 hrs	***	2.17E-4	2.62E-4	1.31E-4
2-8 hrs	***	1.64E-4		6.61E-5
8-24 hrs	***	7.89E-5		4.69E-5
1-4 days	***	4.33E-5		2.23E-5
4-30 days	***	3.35E-5		7.96E-6

***Bounded by the Unit 3 Intake

Note: Current UFSAR value reflects change to correct typo since issuance of Amendment 19.