September 27, 2004

Mr. Karl W. Singer
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3 — ISSUANCE OF AMENDMENTS REGARDING FULL-SCOPE IMPLEMENTATION OF ALTERNATIVE SOURCE TERM (TAC NOS. MB5733, MB5734, MB5735, MC0156, MC0157 AND MC0158) (TS-405)

Dear Mr. Singer:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment Nos. 251, 290, and 249 to Facility Operating License Nos. DPR-33, DPR-52, and DPR-68 for the Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3, respectively. These amendments are in response to your application dated July 31, 2002, as supplemented by letters dated December 9, 2002, February 12, March 26, July 11, and July 17, 2003, and May 17, July 2, August 24, and September 17, 2004.

These amendments adopt the alternative source term (AST) methodology by revising the current accident source term and replacing it with an accident source term as prescribed in Title 10 to the *Code of Federal Regulations* (10 CFR) Section 50.67. The submittals also propose to revise the Technical Specification (TS) sections associated with control room emergency ventilation (CREV), standby gas treatment (SGT), standby liquid control (SLC), and secondary containment systems. Specifically, the amendments modify the licensing and design basis to reflect the application of the AST methodology, the safety-related function of the SLC system, and deletion of a license condition for Units 2 and 3, for which all the actions have been completed. This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the BFN Units 1, 2, and 3 design bases is superseded by the AST proposed by Tennessee Valley Authority (TVA). The previous offsite and control room accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the total effective dose equivalent (TEDE) criteria of 10 CFR 50.67 or small fractions thereof, as defined in Regulatory Guide 1.183. All future radiological analyses performed to demonstrate compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as described in the now-updated BFN Units 1, 2, and 3 design bases.

The February 12, 2003, supplement added TS Section 3.9.9 to require verification that the minimum fuel decay period has passed prior to moving fuel after the reactor is shut down. The July 17, 2003, supplement included the withdrawal of the request to delete one of the TS Sections associated with the absorption of elemental iodine by the SGT and CREV systems charcoal filters.

In a letter dated July 11, 2003, TVA requested an exemption from the requirements of 10 CFR Part 50, Appendix A, General Design Criterion 41, Containment atmosphere cleanup. The NRC staff has determined that the exemption request is not required for the approval for full implementation of AST on the BFN units. However, the information provided in the exemption request was used by the NRC staff to support the enclosed safety evaluation (SE). TVA's implementation of AST methodology requires the addition of a safety-related function to the SLC system. As a result of the new safety-related function for SLC the following license condition will be added:

> The licensee shall maintain the Augmented Quality Program for the Standby Liquid Control system to provide quality control elements to ensure component reliability for the required alternative source term function defined in the Updated Final Safety Analyses Report.

Since these analyses were performed at a power level of 4031 MWt (102 percent of 3952 MWt), the NRC staff finds that the radiological consequences of these design basis accidents would remain bounding up to a rated thermal power of 3952 MWt. However, the approval of this amendment does not constitute authority to operate above the current licensed rated thermal power.

Additionally, the NRC staff reviewed the seismic ruggedness of the structures and components associated with the main steam isolation valve (MSIV) alternative drain path for Unit 1. As a result of this review, the following license condition will be added on Unit 1 only:

> The licensee is required to confirm that the conclusions made in TVA's letter dated September 17, 2004, for the turbine building remain acceptable using seismic demand accelerations based on dynamic seismic analysis prior to the restart of Unit 1.

The approval of this amendment does not constitute a change in the licensing basis of the alternative drain path for Unit 1, it does represent the NRC staff's acceptance of the alternative drain path methodology proposed for use to implement AST. As noted above, TVA must receive approval of the change in licensing and design bases, as well as complete actions necessary to establish a seismically-rugged MSIV leakage alternative drain path, for Unit 1 prior to restart before the BFN Unit 1 loss-of-coolant accident analysis can become effective.

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

Sincerely,

**/RA/**

Eva A. Brown, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-259, 50-260 and 50-296

Enclosures:
1. Amendment No. 251 to License No. DPR-33
2. Amendment No. 290 to License No. DPR-52
3. Amendment No. 249 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures: See next page

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

Sincerely,

**/RA/**

Eva A. Brown, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-259, 50-260 and 50-296

Enclosures: 1. Amendment No. 251 to License No. DPR-33
2. Amendment No. 290 to License No. DPR-52
3. Amendment No. 249 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures: See next page

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| DATE | 9/3/04 | 9/17/04 | 9/ 24/03 | 9/21/04 | ~~09/ 04~~ | 9/24/04 | 9/27/04 |

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cc:
Mr. Ashok S. Bhatnagar, Senior Vice President
Nuclear Operations
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. James E. Maddox, Vice President
Engineering & Technical Services
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Michael D. Skaggs
Site Vice President
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, AL 35609

General Counsel
Tennessee Valley Authority
ET 11A
400 West Summit Hill Drive
Knoxville, TN 37902

Mr. John C. Fornicola, Manager
Nuclear Assurance and Licensing
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Kurt L. Krueger, Plant Manager
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, AL 35609

Mr. Jon R. Rupert, Vice President
Browns Ferry Unit 1 Restart
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, AL 35609

Mr. Robert G. Jones
Browns Ferry Unit 1 Plant Restart Manager
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, AL 35609

Mr. Mark J. Burzynski, Manager
Nuclear Licensing
Tennessee Valley Authority
4X Blue Ridge
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Timothy E. Abney, Manager
Licensing and Industry Affairs
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, AL 35609

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Browns Ferry Nuclear Plant
10833 Shaw Road
Athens, AL 35611-6970

State Health Officer
Alabama Dept. of Public Health
RSA Tower - Administration
Suite 1552
P.O. Box 303017
Montgomery, AL 36130-3017

Chairman
Limestone County Commission
310 West Washington Street
Athens, AL 35611

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 251
License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated July 31, 2002, as supplemented by letters dated December 9, 2002, February 12, March 26, July 11, and July 17, 2003, and May 17, July 2, August 24, and September 17, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Operating License and Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-33 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 251, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Accordingly, the Operating License is amended as indicated in the attachment to this license amendment and subject to the following License Conditions:

   The licensee shall maintain the Augmented Quality Program for the Standby Liquid Control system to provide quality control elements to ensure component reliability for the required alternative source term function defined in the Updated Final Safety Analyses Report.

   The licensee is required to confirm that the conclusions made in TVA's letter dated September 17, 2004, for the turbine building remain acceptable using seismic demand accelerations based on dynamic seismic analysis prior to the restart of Unit 1.

4. This license amendment is effective as of its date of issuance and shall be implemented prior to the restart of Unit 1.

FOR THE NUCLEAR REGULATORY COMMISSION

**/RA/**

Michael L. Marshall, Jr., Acting Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachments: 1. Page of License DPR-33
2. Changes to the Technical Specifications

Date of Issuance: September 27, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 251

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Replace pages 4 and 5 of the Operating License No. DPR-33 with the attached page.

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

| Remove | Insert |
|---|---|
| 3.1-23 | 3.1-23 |
| 3.1-24 | 3.1-24 |
| 3.1-25 | 3.1-25 |
| 3.1-26 | 3.1-26 |
| 3.1-64 | 3.1-64 |
| 3.3-69 | 3.3-69 |
| 3.6-44 | 3.6-44 |
| 3.6-45 | 3.6-45 |
| 3.6-47 | 3.6-47 |
| 3.6-49 | 3.6-49 |
| 3.6-51 | 3.6-51 |
| 3.6-52 | 3.6-52 |
| 3.6-53 | 3.6-53 |
| 3.7-8 | 3.7-8 |
| 3.7-9 | 3.7-9 |
| 3.7-10 | 3.7-10 |
| - - - | 3.9-22 |
| - - - | 3.9-23 |
| - - - | B3.9-36 |
| - - - | B3.9-37 |
| - - - | B3.9-38 |

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 290
License No. DPR-52

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated July 31, 2002, as supplemented by letters dated December 9, 2002, February 12, March 26, July 11, and July 17, 2003, and May 17, July 2, August 24, and September 17, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Operating License and Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 290, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Accordingly, the Operating License is amended as indicated in the attachment to this license amendment and subject to the following License Condition:

   The licensee shall maintain the Augmented Quality Program for the Standby Liquid Control system to provide quality control elements to ensure component reliability for the required alternative source term function defined in the Updated Final Safety Analyses Report.

4. This license amendment is effective as of its date of issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

**/RA/**

Michael L. Marshall, Jr., Acting Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachments: 1. Page of License DPR-52
2. Changes to the Technical Specifications

Date of Issuance: September 27, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 290

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Replace page 6 of Operating License No. DPR-52 with the attached page.

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

| Remove | Insert |
|---|---|
| 3.1-23 | 3.1-23 |
| 3.1-24 | 3.1-24 |
| 3.1-25 | 3.1-25 |
| 3.1-26 | 3.1-26 |
| 3.3-65 | 3.3-65 |
| 3.3-70 | 3.3-70 |
| 3.6-44 | 3.6-44 |
| 3.6-45 | 3.6-45 |
| 3.6-47 | 3.6-47 |
| 3.6-49 | 3.6-49 |
| 3.6-51 | 3.6-51 |
| 3.6-52 | 3.6-52 |
| 3.6-53 | 3.6-53 |
| 3.7-9 | 3.7-9 |
| 3.7-10 | 3.7-10 |
| 3.7-11 | 3.7-11 |
| - - - | 3.9-22 |
| - - - | 3.9-23 |
| - - - | B3.9-36 |
| - - - | B3.9-37 |
| - - - | B3.9-38 |

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 249
License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated July 31, 2002, as supplemented by letters dated December 9, 2002, February 12, March 26, July 11, and July 17, 2003, and May 17, July 2, August 24, and September 17, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Operating License and Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-68 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 249, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Accordingly, the Operating License is amended as indicated in the attachment to this license amendment and subject to the following License Conditions:

   The licensee shall maintain the Augmented Quality Program for the Standby Liquid Control system to provide quality control elements to ensure component reliability for the required alternative source term function defined in the Updated Final Safety Analyses Report.

4. This license amendment is effective as of its date of issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

**/RA/**

Michael L. Marshall, Jr., Acting Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachments: 1. Page of License DPR-68
2. Changes to the Technical Specifications

Date of Issuance: September 27, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 249

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Replace page 5 of Operating License No. DPR-68 with the attached page.

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

| Remove | Insert |
|---|---|
| 3.1-23 | 3.1-23 |
| 3.1-24 | 3.1-24 |
| 3.1-25 | 3.1-25 |
| 3.1-26 | 3.1-26 |
| 3.3-65 | 3.3-65 |
| 3.3-70 | 3.3-70 |
| 3.6-44 | 3.6-44 |
| 3.6-45 | 3.6-45 |
| 3.6-47 | 3.6-47 |
| 3.6-49 | 3.6-49 |
| 3.6-51 | 3.6-51 |
| 3.6-52 | 3.6-52 |
| 3.6-53 | 3.6-53 |
| 3.7-9 | 3.7-9 |
| 3.7-10 | 3.7-10 |
| 3.7-11 | 3.7-11 |
| - - - | 3.9-22 |
| - - - | 3.9-23 |
| - - - | B3.9-36 |
| - - - | B3.9-37 |
| - - - | B3.9-38 |

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 251 TO FACILITY OPERATING LICENSE NO. DPR-33,

AMENDMENT NO. 290 TO FACILITY OPERATING LICENSE NUMBER DPR-52,

AND AMENDMENT NO. 249 TO FACILITY OPERATING LICENSE NUMBER DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3

DOCKET NOS 50-259, 50-260, AND 50-296

## 1.0 INTRODUCTION

By letter to the U. S. Nuclear Regulatory Commission (NRC), dated July 31, 2002, as supplemented by letters dated December 9, 2002, February 12, March 26, July 11, and July 17, 2003, and May 17, July 2, August 24, and September 17, 2004, the Tennessee Valley Authority (TVA, the licensee) submitted a request for an amendment to licenses DPR-33, DPR-52 and DPR-68. These amendments adopt the alternative source term (AST) methodology for Browns Ferry Nuclear (BFN), Units 1, 2, and 3, by revising the current accident source term and replacing it with an accident source term pursuant to Title 10 to the *Code of Federal Regulations* (10 CFR) Section 50.67. The submittals also propose to revise/delete the Technical Specification (TS) sections associated with control room emergency ventilation (CREV), standby gas treatment (SGT), standby liquid control (SLC), and secondary containment systems. Specifically, the amendments modify the licensing and design basis to reflect the application of the AST methodology and the function of the SLC system, and deletion of a license condition for Units 2 and 3, for which all the actions have been completed.

The supplements to the original application include the withdrawal of the request to delete one of the TS sections associated with the absorption of elemental iodine by the SGT and CREV systems charcoal filters. Also, the supplements add a new TS Section to require verification that the minimum fuel decay period has passed prior to moving fuel after the reactor is shut down. TVA had requested deletion of requirements related to the testing of charcoal absorbers in the SGT system (SGTS) and CREV system (CREVS). Although the mitigation capability of the charcoal absorbers in the SGTS and CREVS was not credited in the design basis accident (DBA) radiological consequence analyses, TVA retracted this particular request in a letter dated July 17, 2003. In that letter, TVA stated that the analyses would continue to not take credit for the removal of iodine by the charcoal filters.

In its original submittal, TVA had requested that this request be approved for all three BFN units, but had not submitted complete supporting analyses for Unit 1. TVA had re-analyzed the applicable DBA analyses for Units 2 and 3 and described these in the submittal. The fuel handling accident (FHA) analysis had been performed for all three BFN units. At that time, TVA stated that since the three units were essentially identical, comparable results would be expected for all three units. TVA stated that the existing License Condition 2.C.(4) provided a standing obligation for TVA to submit the remaining analyses for review prior to Unit 1 entering Mode 3 or above. This was affirmed by TVA in its letter of December 9, 2002. By letter dated May 17, 2004, TVA submitted descriptions of the Unit 1 analysis methods, assumptions, inputs, and results, thereby satisfying this obligation.

In a letter dated July 11, 2003, TVA requested an exemption from the requirements of 10 CFR 50, Appendix A, General Design Criterion (GDC) 41, Containment atmosphere cleanup. The NRC staff has determined that the exemption request is not required for the approval for full implementation of AST on the BFN units. However, the information provided in the exemption request was used by the NRC staff to support the technical evaluation discussed below.

The NRC staff reviewed all the supplements. The supplements augmented/withdrew portions of the submittal. The NRC staff determined that although the scope had been modified the originally published no significant hazards consideration determination (67 FR 63697) did not change. However a new *Federal Register* Notice (69 FR 22883) was issued to address the modifications to the submittal not originally noticed.

## 2.0 REGULATORY EVALUATION

In the past, power reactor licensees have typically used U.S. Atomic Energy Commission Technical Information Document (TID)-14844, Calculation of Distance Factors for Power and Test Reactor Sites, dated March 23, 1962, as the basis for DBA analysis source terms. The power reactor siting regulation, which contains offsite dose limits in terms of whole body and thyroid dose, 10 CFR 100.11, Determination of Exclusion Area, Low Population Zone, and Population Center Distance, makes reference to TID-14844.

In December 1999, the NRC issued a new regulation, 10 CFR 50.67, Accident Source Term, which provided a mechanism for licensed power reactors to replace the traditional accident source term used in their DBA analyses with an alternative source term. Section 50.67 of 10 CFR requires a licensee seeking to use an AST to apply for a license amendment and requires that the application contain an evaluation of the consequences of affected DBAs. Regulatory guidance for the implementation of these ASTs is provided in Regulatory Guide (RG) 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors. TVA 's application of July 2002, as supplemented, addresses these requirements in proposing to use the AST described in RG 1.183 as the DBA source term used to evaluate the radiological consequences of DBAs for BFN Units 1, 2 and 3. As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, GDC 19, for the loss-of-coolant accident (LOCA), the main steam line break (MSLB) accident, and the control rod drop accident (CRDA).

Part 50 of 10 CFR, Appendix A, GDC 26, requires that each reactor have two independent reactivity control systems of a different design, while GDC 29 requires that the reactivity control system be capable of accomplishing its safety function in the event of anticipated operational occurrences.

Section 50.49 of 10 CFR, Environmental Qualification of Equipment, requires that the safety-related electrical equipment which are relied upon to remain functional during and following the design basis events be qualified for accident (harsh) environment. This provides assurance that the equipment needed in the event of an accident will perform its intended function. Regulatory Position 1.3.5, 6, and Appendix I of RG 1.189 addresses the requirements for assessing the impact of the difference in source term characteristics on environmental qualification (EQ) doses. NUREG-1465, Accident Source Terms for Light-Water Nuclear Power Plants, provides estimates of AST that were more physically based and that could be applied to a BWR [boiling water reactor]. NUREG-0933 Issue 187, The Potential Impact of Postulated Cesium Concentration on Equipment Qualification1, indicated that for equipment exposed to the containment atmosphere, the TID-14844 source term and the gap and in-vessel releases in the AST produced similar integrated doses, and for equipment exposed to suppression pool water, the integrated doses calculated with the AST remain enveloped by those calculated with TID-14844 for the first 145 days post accident for a BWR, including the 30 percent vs. 1-percent release of cesium. It was concluded that there was no clear basis for back fitting the requirement to modify the design basis for equipment qualification to adopt the AST. There would be no discernible risk reduction associated with such a requirement.

NUREG-800, Standard Review Plan (SRP), Section 6.5.2, Containment Spray as a Fission Product Cleanup System, provides the acceptance criteria regarding the systems used to minimize iodine re-evolution as presented in the licensee's re-analysis of the radiological consequences for the LOCA. The BFN units were not licensed to many of the GDC contained in 10 CFR Part 50, Appendix A, but Section 1.5.1.6 of the Updated Final Safety Analysis Report (UFSAR) contains criteria that are essentially equivalent to GDC's. Maintaining compliance with the intent of these criteria was evaluated as part of the evaluation process.

On March 14, 2003, the NRC staff issued an amendment for Units 2 and 3 to increase the allowable main steam isolation valve (MSIV) leakage rate. This amendment permitted Units 2 and 3 to use the main steam drain lines to direct any MSIV leakage to the main condenser. This drain path takes advantage of the large volume of the main steam lines (MSLs) and condenser to provide holdup and plate-out of fission products that may leak through the closed MSIVs. The licensee performed evaluations and seismic verification walkdowns to demonstrate that the main steam system piping and components which comprise the alternate leakage treatment (ALT) system were seismically rugged and are able to perform the safety function of an MSIV leakage treatment system. By letter dated July 9, 2004, the licensee requested an amendment similar to that granted on Units 2 and 3. The licensee also submitted, in letters dated July 2, August 24, and September 17, 2004, the evaluations, seismic verification walkdowns, and seismic ruggedness evaluations to support the AST use of the ALT MSIV leak path for Unit 1. The seismic ruggedness evaluation was performed to demonstrate the seismic adequacy of the turbine building which houses the ALT system. The structural integrity of the turbine building is an important consideration to the adequacy of the alternate MSIV leakage path because a non-seismically designed turbine building should be capable of withstanding the earthquake without degrading the capability of the ALT system.

The licensee referenced the General Electric Company (GE) Report, NEDC-31858P-A, Boiling Water Reactor Owners Group (BWROG) Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems, Revision (Rev.) 2 (BWROG Report or NEDC-31858P), as a basis for the acceptability of its proposed license amendment. The BWROG report summarizes data on the seismic performance of main steam piping and condensers in past strong-motion earthquakes at various facilities, and compares design attributes of the piping and condensers with those in typical GE Mark I, II, and III nuclear plants. The NRC staff, in its safety evaluation (SE) of the BWROG report dated March 3, 1999, determined that the BWROG approach of utilizing the earthquake experience-based methodology, supplemented by plant-specific seismic adequacy evaluations was not an acceptable basis in and of itself to demonstrate the seismic ruggedness of non-seismically analyzed main steam system piping and condensers. Therefore, the NRC staff identified certain limitations that required individual licensees to provide plant-specific design information and evaluation when BWROG approach was elected for resolving the MSIV leakage issue.

The licensee cited Duane Arnold, Brunswick, Grand Gulf, Hope Creek, Clinton, and Perry as precedents. The NRC staff considers the implementation of an AST to be a significant change to the design basis of the facility that is voluntarily initiated by the licensee. In order to issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding of compliance with regulations applicable to the amendment. During this review, the NRC staff found that the BFN units were not at the same licensing/design bases for these precedents to be of any substantial value. For example, the licensee requested that this amendment be approved for all three BFN units, however, Unit 1 has been shut down since the early 80’s resulting in the need for additional NRC review to resolve issues surrounding design bases verification and required precedent licensing actions.

## 3.0 TECHNICAL EVALUATION

### 3.1 Accident Dose Calculations

The NRC staff reviewed the technical analyses related to the radiological consequences of DBAs that were performed by TVA in support of this proposed license amendment. Information regarding these analyses was provided in the July 21, 2002, submittal and in the supplemental letters dated December 9, 2002, May 17, and July 2, 2004. The NRC staff reviewed the assumptions, inputs, and methods used by TVA to assess these impacts. Independent calculations were performed to confirm the conservatism of the TVA analyses. TVA performed an evaluation of all significant LOCA and non-LOCA events currently analyzed in the BFN UFSAR. These events:

- LOCA
- MSLB
- CRDA
- FHA

For these re-analyses, TVA determined the TEDE at the exclusion area boundary (EAB) for the worst 2-hour period and the 0-30 day low population zone (LPZ) TEDE. TVA also evaluated the potential TEDE to control room personnel from these DBAs. The accident-specific sections that follow describe the accident, the TVA assessment of the impact of the proposed changes, and the NRC staff’s evaluation.

With regard to Unit 1, the inputs used in the MSLB, FHA, and CRDA analyses are bounding for Units 1, 2, and 3. As such, the results determined for these events are applicable for all three units. The inputs used in the LOCA analysis are different for Unit 1 and Units 2 and 3; a separate analysis was performed and the bounding results for Units 1, 2, and 3 were provided. Significant analysis changes are:

- For Units 2 and 3, the turbine building roof ventilator $\chi$/Q values are more conservative than the turbine building exhaust $\chi$/Q values. For Unit 1, the reverse is true. The limiting value for each case was used in the analysis.

- For Units 2 and 3, the reactor building effective mixing free volume is 1,931,502 $ft^3$. For Unit 1, the corresponding volume is taken as 1,311,209 $ft^3$.

- With the exception of the two items above, the remaining LOCA analysis assumptions and inputs are identical, or bounding, for all three units.

3.1.1 Loss-of-Coolant Accident

The objective of analyzing the radiological consequences of a LOCA is to evaluate the performance of various plant safety systems intended to mitigate the postulated release of radioactive materials from the plant to the environment. TVA assumes an abrupt failure of a large reactor coolant pipe and that substantial core damage occurs as a result of this event. The assumption of core damage is conservative in that DBA thermo-hydraulic analyses in the BFN UFSAR conclude the fuel damage thresholds are not exceeded.

3.1.1.1 Source Term

Fission products from the damaged fuel are released into reactor coolant system (RCS) and then into the primary containment (i.e., drywell and wetwell). For a LOCA, it is anticipated that the initial release to the primary containment will last 30 seconds and will release all of the radioactive materials dissolved or suspended in the RCS liquid. The gap inventory release phase begins 2 minutes after the event starts and is assumed to continue for 30 minutes. As the core continues to degrade, the gap inventory release phase ends and the in-vessel release phase begins. This phase continues for 1.5 hours. Tables 1, 4, and 5 of RG 1.183 define the source term used for these two phases. These data are summarized in the attached Table 1. The inventory in each release phase is released at a constant ramp starting at the onset of the phase and continuing over the duration of the phase. Once dispersed in the primary containment, the release to the environment is assumed to occur through five pathways:

- Leakage of primary containment atmosphere (i.e., design leakage).

- Leakage of primary containment atmosphere via design leakage through MSIVs.

- Leakage from emergency core cooling systems (ECCS) that recirculate suppression pool water outside of the primary containment (i.e., design leakage).

- Releases via the containment atmosphere dilution (CAD) system

- Leakage via the hardened wetwell vent (HWWV)

The LOCA considered in this evaluation is a complete and instantaneous severance of one of the recirculation loops. The pipe break results in a blowdown of the reactor pressure vessel (RPV) liquid and steam to the drywell via the severed recirculation pipe. The resulting pressure buildup drives the mixture of steam, water, and other gases down through vents to the downcomers and into the suppression pool water thereby condensing the steam and reducing the pressure. Due to the postulated loss of core cooling, the fuel heats up, resulting in the release of fission products. Under the TID-14844 assumption of instantaneous core damage, this initial blowdown would also include fission products, a fraction of which would be retained by the suppression pool water. Under the AST, the fission product release occurs in phases over a 2-hour period. TVA has conservatively assumed that the fission product release from the RPV is homogeneously dispersed within the drywell free volume only for the first 2 hours. TVA assumes that core quenching occurs at about 2 hours resulting in substantial steam production in the RPV and drywell that will purge a large fraction of the drywell atmosphere through the torus downcomer vents, through the suppression pool, and into the torus airspace. TVA did not credit any reduction in fission products transferred to the torus air space by suppression pool scrubbing. Instead, TVA assumes a well-mixed torus air space and drywell.

TVA assumes that a portion of the fission products released from the RPV will plate-out due to natural deposition processes. TVA models this deposition using the 10-percentile model described in the NRC staff-accepted NUREG/CR-6189, A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments (i.e., the “Powers Model”).

The AST assumes that the iodine released to the containment includes 95 percent CsI, 4.85 percent elemental iodine, and 0.15 percent organic forms. The assumption of this iodine speciation is predicated on maintaining the containment sump water at pH 7.0 or higher. TVA proposes to use the SLC to inject sodium pentaborate (SPB) to the RPV, where it will mix with ECCS flow and spill over to the drywell and then to the suppression pool. SPB, a base, will neutralize acids generated in the post-accident primary containment environment.

3.1.1.2 Containment Leakage Pathway

The drywell and wetwell are projected to leak at their design leakage of 2.0 percent of its contents by weight per day for the 30-day accident duration. Leakage from the drywell and wetwell will collect in the free volume of the secondary containment and be released to the environment via ventilation system exhaust or leakage. Following a LOCA, the SGTS fans start and draw down the secondary containment to create a negative pressure with reference to the environment. This pressure differential ensures that leakage from the drywell and wetwell is collected and processed by the SGTS. SGTS exhaust is processed through HEPA and charcoal filter media prior to release to the environment via the site’s elevated stack. Note that the analyses conservatively did not credit iodine removal by the charcoal filters in the SGTS. TVA assumes that all three SGTS trains are running at the start of the event. TVA states that if only two of the SGTS trains are running, there will be a short period at the start of the event in which the secondary containment may not be at a negative pressure. However, the two SGTS trains can draw the pressure down prior to the onset of the gap release phase at 2 minutes post-accident. The three SGTS train case is conservative as it maximizes the release rate from the secondary containment. A portion of the stack flow is assumed to leak through the backdraft dampers and be released as a ground level release.

3.1.1.3 Main Steam Isolation Valve Leakage

The four MSLs that penetrate the primary containment are automatically isolated by the MSIVs in the event of a LOCA. There are two MSIVs on each steam line, one inside containment (i.e., inboard) and one outside containment (i.e., outboard). The MSIVs are functionally part of the primary containment boundary and design leakage through these valves provides a leakage path for fission products to bypass the secondary containment and enter the environment as a ground level release. TVA conservatively assumes that the fission products released from the core are dispersed equally throughout the drywell via the severed recirculation line. Following the initial blowdown of the RPV, the fuel heats up and fuel melt begins, the steaming in the RPV carries fission products to the drywell. When core cooling is restored, steam is rapidly generated in the core. This steam and the ECCS flow carry fission products from the core to the primary containment via the severed recirculation line, resulting in well-mixed RPV dome and primary containment fission product concentrations. Once the rapid steaming stops, the containment contents can flow back into the RPV through the severed line and would be available for release via the MSIVs.

TVA assumes that one of the four inboard MSIVs fails to close. Therefore, three of the steam lines have a closed space between the inboard and outboard MSIVs; all have the piping volume between the outboard MSIVs and the point at which the drain line path to the condenser connects to the steam line. TVA assumes a maximum MSIV leakage of 100 scfh in the line with the failed inboard MSIV. One of the other lines is assumed to leak at 50 scfh, and the other two lines are assumed not to leak. This modeling is conservative as it minimizes deposition credit. The TVA modeling assumes well-mixed control volumes. Only the piping volumes associated with horizontal runs of MSL piping are included. The amount of fission product aerosol deposition is derived from the methodology in Appendix A to NRC staff report AEB-98-03, Assessment of the Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term. Particulate deposition in the main condenser was treated using the same approach as that for the steam lines. The deposition of elemental iodine in the MSLs is determined using the NRC staff-accepted RADTRAD Bixler model. Since the particulate deposition velocity in the condenser is less than the elemental iodine deposition velocity from SRP 6.5.2, TVA used the particulate deposition velocity.

3.1.1.4 Alternate Leakage Treatment

3.1.1.4.1 Functional Design and Reliability of the Alternate Leakage Treatment Boundary for Unit 1

The BFN alternate leakage treatment (ALT) system for Units 2 and 3 was addressed previously in a license amendment for Units 2 and 3 dated March 14, 2000. TVA submitted an amendment request for Unit 1 on July 9, 2004 (TS-436), the review of which has not been completed. The review in this section addresses the use of the ALT path as it relates to AST for Unit 1.

The ALT utilizes the MSL drains to direct the MSIV leakage to the main condenser. This ALT path takes advantage of the capability of the large volume of the MSLs and condenser to hold-up and plate-out fission products in the MSIV leakage effluent. To mitigate a DBA, this path must be available under DBA conditions with loss-of-offsite-power (LOOP). The ALT path is from the downstream side of the MSIVs through four 3-inch lines which join a 4-inch drain header to the main condenser. In addition to the MSL drains, the drain header also receives drains from high-pressure coolant injection, reactor core isolation coolant steam lines, and the auxiliary boiler. All valves in the flow path are normally open, with the exception of two, FCV-1-58 and FCV-1-59, which are normally closed. FCV-1-59 has a 4-inch bypass line that also routes to the condenser. The bypass around FCV-1-59 is free of valves and orifices; therefore, operation of FCV-1-59 is not essential to align the ALT path.

In the event of an accident, operator actions will establish the primary ALT path to the main condenser. Normally-closed valves FCV-1-58 and FCV-1-59 will be opened using hand switches in the main control room. Both FCV-1-58 and FCV-1-59 will be powered from essential power buses with emergency diesel generator backup. Therefore, these valves are designed to be available during and after a LOCA event concurrent with a LOOP. To further ensure valve reliability, these two valves are in the Inservice Testing Program and will be periodically stroke-tested. The licensee considered the action of including FCV-1-58 in the motor operated valve test programs such as discussed in Generic Letter (GL) 89-10, Safety-Related Motor-Operated Valve Testing and Surveillance, and GL 96-05, Periodic Verification of Design-Basis Capability of Safety-Related Power-Operated Valves. However, the licensee has determined that the American Society of Mechanical Engineers (ASME) Code testing requirements are adequate due to the fact that this valve is not subject to high dynamic loads under the assumed accident conditions.

Additionally, to ensure ALT boundary integrity, the pressure control valve PCV-1-147 on sealing steam supply line will be modified, so that it fails closed instead of open on loss of power, air, or control signal. Also, check valves CKV-1-742 and CKV-1-744 will be added to the steam supply lines to the offgas preheaters to serve as the pressure boundary valves. Subsequently, all of the valves in the ALT boundary will be either: (a) normally closed manual valves, (b) normally closed motor-operated valves, (c) fail-close air-operated valves, or (d) check valves with spring-assisted closure. The licensee has determined that the failure of PCV-1-147 to either an open or closed position results in an operational problem depending on the power level of the reactor, therefore, either state requires operator action. The NRC staff reviewed the operational problems identified by the licensee. As TVA is currently defueled and preparing Unit 1 for restart after almost 19 years, the NRC staff was concerned with the necessary implementation of training and procedures for these modifications. In a letter dated September 17, 2004, the licensee committed to provide training and procedures commensurate with that for Units 2 and 3 for the establishment of the Unit 1 alternate pathway. The NRC staff concludes that based on the modifications to the various valves and the commitment ensuring associated operator actions and training will be performed, the ALT boundary is satisfactory and should not adversely affect normal rector operation.

Section 5.2 of the March 3, 1999, SE states that a secondary path to the condenser, having an orifice, should exist. NEDC-31858P-A does not require that this secondary path have the same flow capability as the primary path. The licensee’s application states that a secondary passive flow path also exists from the MSIVs to the condenser. This secondary path is considered a contingency alignment in the event of the unlikely failure of FCV-1-58. The licensee has determined in the event FCV-1-58 were to fail open, the leakage flow would split, with part of the flow going to the condenser via a 0.1875-inch diameter orifice in a normally open bypass around FCV-1-58, and the remainder going to the condenser via normal leakage paths through the main steam stop/control valves and through the high-pressure turbine. The functional design of the secondary path meets the intent of NEDC-31858P-A, and, therefore, is acceptable.

The NRC staff has reviewed the proposed primary ALT path. Based on the completion of modifications to the Unit 1 valves discussed in this section, revision of procedures and performance of training to address these modifications, the NRC staff concludes that the Unit 1 ALT boundary meets NEDC-31858P-A functional design and reliability criteria, and would be available under post-accident conditions including a LOOP.

3.1.1.4.2 Seismic Walkdown on ALT Pathway

The licensee contracted Facility Risk Consultants (FRC), Inc. to conduct an MSIV seismic ruggedness verification for BFN Unit 1. A report entitled “MSIV Seismic Ruggedness Verification at Browns Ferry Nuclear Plant Unit 1," dated May 2004, was attached to the licensee’s July 2, 2004 letter. The report stated that the BFN Unit-1 MSIV seismic ruggedness verification program was performed in accordance with the recommendations of the GE BWROG Report for increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems, NEDC-31858P, Revision 2, September 1993.

FRC performed a walkdown of the main steam lines, various drain paths, associated components and appendages within the seismic verification boundary for BFN Unit 1 MSIV seismic ruggedness verification program. The walkdown team consisted of four people, all have
college degrees in engineering, and each person possesses ten to twenty years of experience in structural or mechanical engineering and/or earthquake engineering application to nuclear power plants. The team identified 54 potential outliers, which were documented in the Walkdown Data Package. These potential outliers were further evaluated to the acceptance criteria of TVA Design Criteria BFN-50-C-7306, “Qualification Criteria for Seismic Class II Piping, Pipe Supports, and Components.” Further evaluation utilized hand calculations for simple piping configurations and rigorous piping analysis (TPIPE computer program) for complex piping configurations. A total of 15 outliers were found to have not met the acceptance criteria. Plant modifications were designed and several Design Change Notices were issued to implement changes determined necessary to resolve identified outliers. Furthermore, 15 maintenance and/or housekeeping items were also identified for corrective actions. Maintenance work order requests were issued to address these items. The walkdown performed by FRC, in accordance with the procedures of the BWROG NEDC-31858P, Revision 2, September 1993, is acceptable to the staff.

The pipe support acceptance criteria in the FRC report accept support material to go beyond yield for non-ductile behavior supports. The staff in a letter, dated September 14, 2004, requested the licensee to provide justification for such a criterion. In the September 17, 2004, submittal, the licensee stated that, after a review, it had not used supports constructed from non-ductile materials. The staff also requested the licensee to provide justification for accepting loads on test data with mean less one standard deviation capacity. The licensee responded that it had not used such a criterion for Unit 1. The staff further requested the licensee to provide justification for considering those pipe supports that failed the stress criteria to be acceptable if their adjacent supports and the resulting pipe span can resist dead loads with a factor of safety of 2.0. The licensee responded that, after a review, Unit 1 does not have such a condition. The staff considers the licensee’s responses satisfactory to resolve the requests for additional information (RAIs).

3.1.1.4.3  Condenser

The FRC report stated that the main condenser anchorage was reviewed during the walkdown. Each of the three condensers is mounted on concrete pedestals. The concrete pedestals were observed to be in good condition. The report further stated that confirmation of the condenser seismic capacity and anchorage adequacy is required to ensure its structural integrity during a design-basis earthquake (DBE) seismic event. The licensee submitted additional information on the condenser in an August 24, 2004 letter. The licensee stated that the condensers were analyzed for structural integrity to seismic DBE load. Results of the analysis indicate that the condenser shell stresses are relatively small for both combined axial and bending, and shear.

The condenser support anchorage consists of a center key and six support feet. The center support is a fixed anchor and consists of a built-up wide flange H section embedded 4 feet into the concrete pedestal, which is connected to the turbine building base mat and welded to the bottom plate of the condenser. The support plates consist of two to three anchors of 2- to 2-1/2-inch diameter bolts. Each anchor bolt has greater than 5-feet nominal length with approximately 48-inches of embedment into the concrete pedestal, which is connected to the turbine building base mat. These supports were designed to resist vertical operating loads and are slotted radially from the center key to allow for thermal growth. Shear forces are transferred to the H-shaped anchor in the center, to the anchor bolts and shear keys to the support feet, and carried through the concrete pedestal to the turbine building base mat. The anchorage for the BFN condenser is comparable with the performance of the anchorages for similar condensers in the earthquake experience database. The shear areas of the condenser anchorage, in the directions parallel and transverse to the turbine generator axis, divided by the seismic demand, were used to compare with those presented in the NEDC-31858P-A report. The BFN condenser anchorage shear area to seismic demand is greater than those in the selected database sites. The condenser support anchorage was also evaluated and the results indicate that the combined seismic DBE and operational demand are less than the anchorage capacity based on the American Institute of Steel Construction (AISC) allowables. Maximum stress ratios are 0.57 for bolt tension in the perimeter support feet and 0.95 for shear in the center support built-up section.

The staff finds the licensee use of seismic analysis for the condenser, and the NEDC-31858P-A data for evaluating the condenser anchorage capacity acceptable.

3.1.1.4.4  Turbine Building

The licensee's August 24, 2004, submittal on the turbine building concluded that the turbine building would remain intact based on earthquake experience data. The referenced experience data was summarized in a statement that, ". . . there are no known cases of structural collapse of either turbine buildings at power plant stations or structures of a similar construction." In an RAI, dated September 14, 2004, the staff informed the licensee that the staff had not endorsed the use of seismic experience data for qualifying structures subjected to earthquakes, and requested additional technical justification that supports the contention that the BFN Unit 1 turbine building will remain structurally intact following a DBE.

The licensee's September 17, 2004, letter stated that it had performed an evaluation of the turbine building seismic capacity against seismic demand generated by a DBE. The evaluation includes the lower reinforced concrete frame and shear wall structure, turbine pedestal, and the steel superstructure. The staff evaluation results are stated below.

Lower Reinforced Concrete Frame and Shear Wall Structure: The licensee calculated the capacity of concrete wall and column at a cross section believed to have the least seismic capacity coupled with a great seismic demand. The calculated seismic capacity of the shear walls and columns is 25,500 kips in the north-south direction and 20,300 kips in the east-west direction. The shear capacity was calculated based on nominal cross sectional properties of the walls and columns and concrete shear strength as specified in the American Concrete Institute 318-89 Code. The licensee also estimated the seismic demand at the same section. The licensee used an amplification factor of 1.6 for soil-founded Class I structures, as stated in the UFSAR, and calculated the seismic demand spectral acceleration using 1.6 times the 0.30g peak of the 5 percent damped DBE ground response spectrum curve. The seismic demand shear load is calculated by the total dead load plus the design live load above the cross section times the seismic demand spectral acceleration, which equals 15,200 kips for both the north-south and east-west directions of earthquake motion. The calculated seismic capacity over demand (C/D) ratios for the lower concrete structure is 1.68 for the north-south direction and 1.34 for the east-west direction. Since the seismic capacities are greater than the seismic demand, the lower reinforced concrete frame and shear wall structure will remain intact following a DBE.

Turbine Pedestal: The turbine pedestal is a separate reinforced concrete structure housed within the turbine building. The turbine pedestal was previously evaluated for lateral loads equivalent to 25 percent of the weight of the turbine generator machinery, applied in both lateral directions. The evaluation was described in the UFSAR. The evaluation used working (allowable) stress method. Therefore, the seismic capacity of the turbine pedestal was calculated to be equal to 0.425g (1.7 x 0.25g). Since the turbine pedestal is a rigid structure, its seismic demand equals the peak ground acceleration of the ground motion DBE (0.20g), increased by 1.6 to account for soil amplification effects, as described in the UFSAR. This yields a seismic demand acceleration of 0.32g (1.6 x 0.2g). The seismic C/D ratio for the turbine pedestal is 1.33. Since the seismic capacity of the turbine pedestal is greater than the demand generated by the DBE, the turbine pedestal will remain intact following a DBE.

Steel Superstructure: The Turbine Building consists of eight (8) two-span high-bay moment resisting frames in the east-west direction, and braced frames in the north-south direction. The original design of the steel superstructure was based on dead and live loads, plus loading due to 100 mph wind (30 psf lateral load on the entire structure) and lift loads for the turbine building crane. As a seismic II/I verification, a typical moment frame in the east-west direction, a braced frame in the north-south direction, and a simple roof girder in the vertical direction are used. The analysis assumes that the frames in the east-west and north-south directions behave as a single-degree-of-freedom oscillator and that the mass on the roof girder is uniformly distributed. Standard structural mechanics analysis methodology was used to determine the stiffness and strength of the east-west and north-south load resisting frames for lateral loads applied at the roof line of the superstructure. Mass was lumped at the roof line. The mass was determined based on the weight of the roof framing of the structural frame, the tributary weight of the roofing, and ½ of the weight of the columns, longitudinal framing (including the crane rail), and the siding of the building.

The capacity of structural members and connections were calculated based on Part 2 of the AISC specification. The limiting condition that governs the capacity of the steel superstructure in the east-west direction was determined to be the bending capacity at the top of the main support columns, and the calculated capacity is 73,000 in-kips. The limiting condition that governs the capacity in the north-south direction was determined to be the capacity of the connections for the brace members, and the calculated capacity is 311 kips. The limiting condition that governs the capacity in the vertical direction of the roof girder was determined to be the bending strength at the center of the roof girder, and the calculated capacity is 38,200 in-kips.

Seismic demand loads were determined by dead load plus seismic spectral acceleration times the mass. The seismic load cases investigated include a DBE in the east-west direction plus vertical direction, and in the north-south direction plus vertical direction. The spectral accelerations used for the evaluation are taken from an approximation of the floor response spectra at the El. 617 ft operating deck. The floor response spectra were approximated based on scaling from the DBE ground motion response spectrum. The scale factor for horizontal direction motion is taken as 1.6 x 1.5 x 1.5 = 3.6. As described above, the 1.6 coefficient was used to represent the soil amplification. The first 1.5 factor is to account for building amplification up to an elevation of 40 ft above grade, based on the Seismic Qualification Utility Group General Implementation Procedure, up to El. 605 ft. The second 1.5 amplification factor was used to account for additional building amplification from El. 605 ft to the operating deck level of El. 617 ft. In the vertical direction, the scale factor was taken as 2/3 x 1.1 = 0.733. The 2/3 factor is the ratio between vertical and horizontal ground motion as defined in the UFSAR. The 1.1 coefficient is for soil amplification in the vertical direction, as described in the UFSAR for soil-founded Class I structures. The spectral acceleration values applied for the seismic load analysis were taken from the 5 percent damped floor response spectra at the natural frequency of the structure. Frequency was calculated using the respective stiffness and mass for the east-west and north-south mathematical models. The calculated natural frequency is 1.13 Hz for the frame in the east-west direction, and 2.74 Hz for the frame in the north-south direction. The natural frequency of the roof girder in the vertical direction was determined using the 1g deflection approximation methodology. The calculated natural frequency is 2.34 Hz for the roof girder in the vertical direction. The calculated seismic demand for the frame in the east-west direction is 48,500 in-kips at the top of the column and 15,500 in-kips at the center of the roof girder. The calculate seismic demand for the frame in the North-South direction is 299 kips shear force.

The C/D ratios for the steel superstructure are 1.51 for the frame in the east-west direction, 1.11 for the frame in the north-south direction, and 2.47 for the roof girder in the vertical direction. Since the resulting C/D ratios are greater than 1.0, the steel superstructure was determined to remain intact following a DBE.

The staff did not review seismic experience-based seismic II/I verification of the turbine building, because it has not endorsed such a method for structures. The staff concurs with the licensee’s methods and assumptions for calculating seismic capacity of the turbine building. The staff finds that the licensee’s methods and assumptions for calculating seismic demands for the turbine building subject to a DBE are approximate and reasonable. Therefore, the staff concludes that the licensee has demonstrated, with some confidence, that the turbine building will remain intact following a DBE event. Since the capacity/demand ratio is as low as 1.11 for the frame in the north-south direction, and several approximations were used both in the assumptions and hand-calculation methods, the NRC staff requires the addition of the following license condition:

> The licensee is required to confirm that the conclusions made in TVA's letter dated September 17, 2004, for the turbine building remain acceptable using seismic demand accelerations based on dynamic seismic analysis prior to the restart of Unit 1.

Based on the above evaluation, the NRC staff concludes there is reasonable assurance that the BFN Unit 1 MSIV ALT system is seismically adequate for the intended purpose. The staff approval of this amendment is contingent on licensee's confirmation of satisfying the license condition to perform a dynamic analysis for the turbine building as discussed above. The staff's conclusion is based on the fact that (1) the ALT pathway has been walkdown in accordance with the procedures in the NEDC-31858P-A report, which was approved by the NRC staff, (2) all the outliers have been either analytically resolved or physically modified, (3) the condenser was seismically analyzed subject to a DBE for its adequacy and its anchorages were evaluated to be adequate in accordance with information contained in the NEDC-31858P-A report, and (4) the turbine building is deemed, through the use of approximate calculations to be followed by performance of dynamic seismic analysis prior to the plant restart, to remain intact following a DBE.

It should be noted that the staff's acceptance of the experience-based methodology as presented by the BWROG and BFN Unit 1, is restricted to its application for ensuring the pressure boundary integrity and functionality of the alternate drain pathway associated with the MSIV leakage treatment system. The staff's acceptance of the methodology for this application is not an endorsement for the use of the experience-based methodology for other applications at BFN Unit 1.

3.1.1.5 Leakage from Emergency Core Cooling Systems

During the progression of a LOCA, some fission products released from the fuel will be carried to the suppression pool via spillage from the RCS and by natural processes such as deposition and plate-out. Post-LOCA, the suppression pool is a source of water for ECCS. Since portions of these systems are located outside of the primary containment, leakage from these systems is evaluated as a potential radiation exposure pathway. For the purposes of assessing the consequences of leakage from the ECCS, TVA assumes that all of the radioiodines released from the fuel are instantaneously moved to the suppression pool. Noble gases are assumed to remain in the drywell atmosphere. Since aerosols and particulate radionuclides will not become airborne on release from the ECCS, they are not included in the ECCS source term. This source term assumption is conservative, in that all of the radioiodine released from the fuel is credited in both the primary containment atmosphere leakage and the ECCS leakage. In a mechanistic treatment, the radioiodines in the primary containment atmosphere would relocate to the suppression pool over time.

The analysis considers the equivalent of 5 gallons per minute (gpm) unfiltered ECCS leakage starting at the onset of the LOCA. TVA assumes the 10 percent of the iodine in the ECCS leakage becomes airborne and is available for release as 97 percent elemental and 3 percent organic iodine. No credit was assumed for hold-up and dilution in the secondary containment.

As assumed for the primary containment leakage pathway, the leakage enters the environment via the SGTS as a filtered elevated release. The release continues for 30 days.

3.1.1.6 Other Release Paths

TVA models two additional release paths that vent the torus air space to the stack. The first is the CAD system. This system is operated for 24 hours at 10, 20, and 29 days post-accident. The flow rate from this system, which is directed through the SGTS filters, is 139 cfm. The second path is the HWWV which is postulated to leak at a rate of 10 cfm starting at 8 hours and continuing for 30 days. This latter release does not pass through the SGTS filters.

3.1.1.7 Offsite Doses

TVA evaluated the maximum 2-hour TEDE to an individual located at the EAB and the 30-day TEDE to an individual at the outer boundary of the LPZ. The resulting doses are less than the 10 CFR 50.67 criteria.

3.1.1.8 Control Room Doses

TVA evaluated the dose to the operators in the control room. The CREVS is automatically actuated by a primary containment isolation signal (PCIS), by high radiation at the control bay air intakes, or by manual actuation by the control room operators. Since the PCIS actuation is triggered by plant process sensors, such as RPV low water level and high drywell pressure, isolation of the control room is assumed to be immediate (i.e., completed before substantial fission products are released) for the DBA LOCA. Although TVA stated in Table 2-10 of its submittal that CREVS would enter the pressurization mode for the MSLB and the CRDA, no credit is taken for filtration by HEPA filters or charcoal media for these events. Initially, the control room ventilation system intake is 3000 cfm plus 3717 cfm of unfiltered inleakage. Once isolation occurs, the 3000 cfm intake is filtered and the 3717 cfm of unfiltered inleakage continues.

Although the control room is designed to be pressurized during an accident event, TVA assumes that unfiltered inleakage into the control building habitability zone (CBHZ) occurs. In May 1992, TVA performed testing and determined the unfiltered inleakage rate was 3717 cfm. Additional testing, performed using special test fans to maintain a positive pressure of 0.50-inch water gauge, determined an inleakage rate of 3815 cfm. The results obtained using the CREVS fans showed an inleakage rate of 3189 cfm. This additional testing was described in a TVA letter dated May 18, 1993. For this amendment request, the NRC staff requested additional information with regard to actions taken by TVA to establish the continued validity of the assumed unfiltered inleakage value. In its response dated December 9, 2002, TVA stated that the assumed unfiltered inleakage value is validated by surveillance testing every 24 months. TVA also stated that it has administrative programs in place to control penetrations into the CBHZ and to maintain door seals.

The NRC staff is currently working toward resolution of generic issues related to control room habitability, in particular, the validity of control room inleakage rates assumed by licensees in analyses of control room habitability. The NRC staff issued GL 2003-01, Control Room Habitability. TVA responded to this GL by letter dated December 8, 2003. In this response, TVA reported that inleakage testing using the American Society for Testing Materials tracer gas methodology yielded a control room unfiltered inleakage rate of only 600 cfm. This value is approximately 84 percent less than the 3717 cfm assumed in the BFN design and licensing basis, a conservative situation. Although the TVA response to the generic letter is still under review, the NRC staff has determined that there is reasonable assurance that the BFN control room will be habitable during DBAs and that this amendment may be approved before the final resolution of the generic issue. The NRC staff bases this determination on (1) the results of the tracer gas testing, (2) the relative magnitude of the infiltration currently assumed in the BFN analyses, and (3) favorable site $\chi$/Q values. The NRC staff's acceptance of TVA's unfiltered inleakage assumption for the purpose of this amendment request does not establish that the NRC staff has found the December 8, 2003 response adequate. The NRC staff will respond to TVA's generic letter response under separate cover once its review is complete.

The assumptions found acceptable to the NRC staff are presented in Tables 1 and 2. The EAB, LPZ, and control room doses estimated by TVA for the LOCA were found to be acceptable. The NRC staff performed independent calculations and confirmed the TVA conclusions.

3.1.1.9 Suppression Pool Post-LOCA pH

One of the modifications specified in the submittal is a requirement for maintaining basic pH in the suppression pool in order to minimize release of the radioactive iodine. The licensee developed a method for controlling pH by using the buffering action of SPB released to the suppression pool from the SLC system (SLCS). The amount of the SPB needed to maintain basic pH in the suppression pool was calculated using a computer program. The NRC staff reviewed this calculation and concurs with the licensee that buffering action of SPB will ensure that the suppression pool's pH will stay above 7 for the period of 30 days after beginning of the accident.

Section 5.1.2, Credit for Engineered Safety Features, of RG 1.183 provides the criteria for safety-related features that provide accident mitigation. These criteria include TS operability, powered by emergency power sources, and are actuated automatically or are actuated in accordance with emergency operating procedures. Additionally, the single active component failure that results in the most limiting radiological consequences should be assumed. As discussed previously, the licensee elected to use the SLCS to minimize the release of radioactive iodine. The NRC staff reviewed the licensee's evaluation regarding the use of the SLCS for the safety-related function. The NRC staff found that the SLCS failed to meet all the requirements of a safety-related system in that SLCS is not designed for the single active component failure, nor has the system been procured consistent with the requirements for safety-related systems. To provide reasonable assurance regarding the reliability and redundancy of the system the licensee designated the following design, inspection and programs:

a. The system has seismic Class 1 design of components required for reactivity control and new suppression pool pH control functions.

b. The system is provided with standby AC power supplemented by the emergency diesel generators.

c. The system is subject ASME Section XI, Inservice Inspection requirements.

d. The system is within the scope of the BFN 10 CFR 50.65 Maintenance Rule Program.

e. Most components (pumps, squib valves, etc.) are redundant in parallel trains powered from different electrical busses. The exceptions are the containment isolation check valves and the selector switch in the main control room.

f. Procedures will be updated to activate the SLC system in two hours post-LOCA. The activation will be based on fuel failure as determined by high radiation in the primary containment.

g. Training will be provided on the new SLC injection function during operator requalification training.

Although not completely meeting the single failure criteria, the NRC staff reviewed the components that could be subject to single failure. Two components were identified, the containment isolation check valves and the main control room selector switch. The containment isolation valves are 1 ½ inch Velan stainless steel piston check valves procured under ASME, Section 3, Class 2 design requirements as safety related due to the containment isolation function. In the periodic inspections and testing of these valves, BFN has not experienced any failures of these valves or similar valves on pump discharge. Industry databases (EPIX and NPRDS) confirm that no failures to open or close have been reported on valves of this manufacture and type. Although acknowledging that a single failure to open of one of the two check valves could prevent SLC injection, the NRC staff has determined that the potential for failure is very low based on the quality as established by its procurement as an ASME, Section 3, Class 2 safety-related valve, periodic testing and inspection, and historical performance of the component.

The NRC staff also acknowledged that the selector switch in the main control room could fail and prevent either train or both trains of injection from functioning. The NRC staff determined that the switch was a high reliable component at an accessible location. The switch could easily be replaced or bypassed to start one of the SLC trains if the switch were to fail.

The NRC staff considered the transport of the SPB from the reactor vessel to the suppression pool. The SLC system injects the SPB to the reactor vessel. The transport of reactor vessel contents including the SPB to the suppression pool is by flow through the break (assumed to be a large recirculation pipe break) to the drains that feed the suppression pool. The licensee stated that the Core Spray (CS) injection would provide water directly to the core. One train of flow would be 5600 gallons with a maximum of 600 gallons being lost to steaming. The water would flow downward in the core to the bottom of jet pumps and then flow upward in the jet pumps and out the break. This flow would sweep the SPB from the vessel to the suppression pool. The licensee provided a cross section of the reactor vessel showing the SLC injection location and the flow path for the CS injection that mixes with the SPB and transports it from the vessel. The large CS flow combined with the relatively small SLC injection flow provides good mixing and a significant level of transport from the vessel.

The licensee also stated that the residual heat removal (RHR) system operating in the cooling mode would provide mixing of the suppression pool. The NRC staff concluded that there would be mixing and transport at some rate and that it was reasonable to assume the concentration of SPB in the core would equalize with the concentration in the suppression pool within an acceptable time after SLC injection. As a consequence, there would be sufficient pH control to deter and prevent iodine re-evolution.

The specific changes being made to TS 3.1.7 are (1) the extension of applicability to Mode 3, (2) an additional Required Action and Completion Time for Action C, (3) a change in Surveillance Requirement (SR) 3.1.7.1 increasing the volume of the SPB solution from 3007 gallons to 4000 gallons, and (4) an additional surveillance requirement, SR 3.1.7.3, which verifies the SPB concentration and specifies the frequency testing.

Most of the calculations of the suppression pool pH, presented in the submittal, were performed using a proprietary computer code not accessible to the NRC staff. However, the licensee provided enough information to permit the NRC staff to perform its independent verification. After a LOCA, several acidic species are introduced into the suppression pool. The main sources of acidic species are hydrochloric and nitric acids. Hydrochloric acid is generated by decomposition of the Hypalon and PVC cable insulation. Only the insulation on the cables exposed directly to radiation field is decomposed. Significant portion of cables remains shielded from radiation by metal conduits and they are not significantly affected by radiation. Nitric acid is produced by irradiation of water and air in the radiation environment existing in the containment after a LOCA. The only significant source of basic species is cesium hydroxide released from the damage fuel. With these chemical species and without buffering action of SPB, the pH in the suppression pool water will drop below 7 in about 1 day. However, with a sufficient amount of buffer, the pH in the suppression pool could be maintained above 7 for 30 days.

The amount of SPB required for maintaining sufficient buffering capacity was calculated by a contractor using the STARpH computer code. This code has been used previously by other licensees for determining the required amount of SPB buffers. For the BFN, 4000 gallons of 8-weight percent of SPB solution was required to provide sufficient amount of buffer. This solution was provided from the plant's SLCS. Although the primary function of the SLCS is to introduce negative reactivity to the core, in the event of a control rod mechanism failure, it was extended to include injecting SPB solution into the suppression pool for pH control. The qualification of the SLCS to perform this function in a post-LOCA environment was verified by the NRC staff. The AST analysis specifies manual initiation of SLCS 2 hours after the beginning of the accident and its completion when adequate volume of SPB solution is introduced into the suppression pool. The current plant TSs will be revised to increase the minimum required volume of SPB solution from 3007 to 4000 gallons, and to verify that the concentration of SPB in the SLCS is equal or higher than 8-weight percent. Subsequently, the licensee found that the amount of the cable which was a source of hydrochloric acid was underestimated in the calculation of pH. The calculation based on a corrected value of the cable was submitted in a letter dated May 7, 2004. These calculations were done for Units 2 and 3, and separately for Unit 1 because the amount of acid-generating cable in Unit 1 is different. The results of the modified calculations did not change the final conclusion of the SLCS control of sump pH.

In the submittal, the licensee described its methodology for controlling the post-LOCA pH in the suppression pool water above 7. The methodology relies on using buffering action of SPB, introduced into the suppression pool from the SLCS. The licensee provided analyses justifying that 4000 gallons of 8-weight percent solution of SPB will ensure that pH in the suppression pool will stay above 7 for 30 days after a LOCA. The NRC staff has reviewed the analysis and justifications provided by the licensee and concludes that the analysis presented in the licensee's submittal indicates that the suppression pool pH will stay basic (above 7) for the period of 30 days after a LOCA.

The licensee indicated that the existing Augmented Quality Program contains stringent controls to provide adequate quality control elements such as procurement of replacement parts and control of maintenance activities to ensure the reliability of the SLCS. From the licensee statements, the NRC staff has concluded that although not designated safety related, the SLCS as installed at BFN is a system of sufficient quality that provides reasonable assurance that the SPB will be injected into the core upon activation. However to ensure the reliability, redundancy, and quality of the system is maintained the following condition is being added to the license:

> The licensee shall maintain the Augmented Quality Program for the Standby Liquid Control system to provide quality control elements to ensure component reliability for the required alternative source term function defined in the Updated Final Safety Analyses Report.

Based on the SLCS having suitable redundancy, in components and features, suitable interconnections, leak detection, isolation, and containment capabilities, adequate emergency power available to ensure it's safety function can be accomplished, the NRC staff finds the use of SLCS to minimize the re-evolution of iodine into the containment atmosphere is acceptable. The assumptions regarding the dose consequences for a LOCA found acceptable by the NRC staff are presented in Tables 1 and 2. The NRC staff performed independent calculations and confirmed the TVA conclusions.

3.1.2 Main Steam Line Break

The accident considered is the complete severance of an MSL outside the primary containment. The radiological consequences of a break outside containment will bound the results from a break inside containment. No fuel damage is projected to occur. The MSIVs are assumed to isolate the leak within 5.5 seconds. No other release mitigation is assumed. The analysis is performed for two activity release cases, based on the maximum equilibrium and pre-accident iodine spike concentrations of 3.2 uCi/gm and 32.0 uCi/gm dose equivalent I-131, respectively. TVA assumes that the iodine specie in the release is entirely elemental. This assumption differs from that in RG 1.183. However, the difference is inconsequential since no credit was taken for filtration or other iodine removal mechanisms. The control room was modeled as described above for the LOCA, with the exception that no credit is taken for CREVS filters or charcoal beds.

The assumptions found acceptable to the NRC staff are presented in Tables 1 and 2. The EAB, LPZ, and control room doses estimated by TVA for the MSLB were found to be acceptable. The NRC staff performed independent calculations and confirmed the TVA conclusions.

3.1.3 Control Rod Drop Accident

This accident analysis postulates a sequence of mechanical failures that result in the rapid removal (i.e., drop) of a control rod. A reactor trip will occur. Localized damage to fuel cladding and a limited amount of fuel melt are projected. The MSIVs are assumed to remain open for the duration of the event. TVA has projected 850 fuel rods would be breached by the event and, of these damaged rods, 0.77 percent would exceed the threshold for melting. The analysis assumes that the fission products released from the damaged fuel are instantaneously transported to the main condenser. It is assumed that 100 percent of the noble gases and 10 percent of the iodines released reach the main condenser due to plate-out in the RPV and main steam lines. Of the iodine that enters the main condenser, 90 percent plates out. These assumptions are consistent with the guidance of RG 1.183. There is no reduction in noble gases. It is further assumed that the main condenser is evacuated for 30 days after the event by either the steam jet air ejectors or the mechanical vacuum pumps. The mechanical vacuum pump release path is used for this analysis as this results in the most limiting fission product release. The mechanical vacuum pump discharges via the stack and enters the environment as an elevated release. A portion of the stack flow is assumed to leak through the backdraft dampers and be released as a ground level release. This transport path differs from the guidance in RG 1.183, which states that the release from the condenser should be modeled as 1.0 percent per day for 24 hours as a ground level release. The NRC staff considers this to be an acceptable alternative in that the mechanical vacuum pump flow yields an effective release rate of 1425 percent per day for 30 days, which is more limiting. The current licensing basis analysis assumed two additional release paths. In response to an NRC staff request for additional information, TVA stated that these pathways had been considered, but the mechanical vacuum pump case was more limiting under the original source term and the proposed AST. The control room was modeled as described above for the LOCA, with the exception that no credit is taken for CREVS filters or charcoal beds.

The assumptions found acceptable to the NRC staff are presented in Tables 1 and 2. The EAB, LPZ, and control room doses estimated by TVA for the CRDA were found to be acceptable. The NRC staff performed independent calculations and confirmed the TVA conclusions.

3.1.4 Fuel Handling Accident

This accident analysis postulates that a spent fuel assembly is dropped during refueling. The kinetic energy developed in this drop is conservatively assumed to be dissipated in the damage to the cladding on 111 fuel rods. The fission product inventory in the core is largely contained in the fuel pellets that are enclosed in the fuel rod clad. However, the volatile constituents of this inventory will migrate from the pellets to the gap between the pellets and the fuel rod clad. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released because of the accident. Fission products released from the damaged fuel are decontaminated by passage through the pool water, depending on their physical and chemical form. TVA assumed no decontamination for noble gases, a factor of 200 decontamination of radioiodines, and retention of all aerosol and particulate fission products. The fission products released from the pool are assumed to be released to the environment without credit for reactor building holdup or dilution, via the refueling building ventilation system. The control room was modeled as described above for the LOCA, with the exception that no credit is taken for CREVS filters or charcoal beds.

The assumptions found acceptable to the NRC staff are presented in Tables 1 and 2. The EAB, LPZ, and control room doses estimated by TVA for the FHA were found to be acceptable. The NRC staff performed independent calculations and confirmed the TVA conclusions.

### 3.2 Atmospheric Dispersion

The $\chi$/Q values used in this assessment were developed prior to the current amendment and were previously accepted by the NRC staff for Amendment Nos. 263 and 223 for BFN Units 2 and 3. The exhaust ventilation systems at BFN are common to all three units. As such, the same release points are generally involved with the accident sequences regardless of which unit is experiencing the event. There are a common Unit 1 and 2 control room and a separate Unit 3 control room. Both control rooms are served by a common ventilation system that has two separated outside air intakes. These intakes, although designated as “Unit 1” or “Unit 3,” serve both control rooms concurrently. The ductwork and damper configuration does not provide the capability to isolate either intake from either control room, but does provide for mixing of the two air streams. In previous analyses for Units 2 and 3, TVA utilized the more limiting $\chi$/Q value for the control room intakes divided by two to account for dilution from air drawn into the other intake as provided for in regulatory guidance. There are two release points for the turbine building–the turbine building roof ventilators and the turbine building exhaust. For Unit 2 and 3, the roof ventilator values were more conservative and were used in analyses of the post-LOCA MSIV leakage consequences. However, for Unit 1, the turbine building exhaust is more limiting. As such, the Unit 1 LOCA utilized the turbine building exhaust $\chi$/Q values in assessing MSIV leakage.

TVA proposed a new instantaneous puff release $\chi$/Q value for the control room based on a new, unreviewed dispersion methodology. TVA proposed to use this value as a replacement for values in the current BFN licensing basis. In Draft Guide DG-1111, Atmospheric Relative Concentrations for Control Room Habitability Assessments at Nuclear Power Plants, the NRC staff proposed an acceptable model for puff release $\chi$/Q values. That model was derived from empirically-based meteorological dispersion methods. This guide was issued for public comment in December 2001 (subsequently issued as RG 1.194 in June 2003). The puff dispersion model proposed by TVA is not derived from standard meteorological dispersion formulations but rather is based largely on thermodynamic phenomena. TVA provided no empirical confirmation that its proposed methodology was appropriate. In response, TVA elected to continue to use the current BFN licensing basis value and retract the proposed new methodology. Since the proposed new $\chi$/Q value was slightly greater than the current licensing basis value, the docketed dose results of the MSLB analysis performed by TVA are acceptable for the present amendment request.

### 3.3 Proposed Technical Specification Changes

The licensee has proposed the following changes to the BFN TSs for Units 1, 2, and 3. The changes evaluated below are considered to be design changes to the facility. The TS requirements that are being deleted have not been identified as being relocated to other documents, such that the control provided by the requirements will be completely removed from the facility.

3.3.1 TS 3.1.7, Standby Liquid Control (SLC) System

TS 3.1.7 is being changed to revise the required amount of SPB from ~3007 gallons to ~4000 gallons. Additionally, a new surveillance requirement to verify that the SPB concentration is ~8.0 percent 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 LOCA involving fuel damage.

TVA requested three changes to TS 3.1.7. The first change is to Limiting Condition for Operation (LCO) 3.1.7, which will add the requirement that the reactor be in Mode 4 within 36 hours if the required action and associated completion time are not met. The second change is to increase the required volume of SPB from greater than 3007 gallons to greater than 4000 gallons in SR 3.1.7.1. The third change is to add SR 3.1.7.3, which requires that verification of the SPB concentration above 8.0 percent by weight be performed every 31 days and once within 24 hours after water or boron is added to solution.

The requirement in LCO 3.1.7 for the reactor to be in Mode 4 in 36 hours if the required action and associated Completion Time is not needed because after implementation of AST, the SLC system operability will be required in Mode 3. This addition does not affect the previous requirements for reactor shutdown based on SLC system availability and increases the shutdown requirements. Therefore, the NRC staff finds this additional shutdown requirement to LCO 3.1.7 acceptable.

The modification to SR 3.1.7.1 will increase the volume of SPB solution to greater than 4000 gallons. Since the proposed SR 3.1.7.3 and SR 3.1.7.4 maintain the concentration of the SPB within a range between 8.0 percent and 9.2 percent by weight and SR 3.1.7.6 maintains the SLC parameters for meeting Anticipated Transient Without Scram (ATWS) concerns, changing the volume of the solution within the tank will not impact the concentration of solution entering the reactor in the event of an accident. Similarly, SR 3.1.7.7 specifies that the pump flow rate must be greater than 39 gpm and no change is proposed to this SR; therefore, the rate that the solution enters the reactor will not change. Therefore, the reactivity control function of the SLC system will not be impacted by the proposed change and the SLC system will continue to meet GDC 26 and 29, so changing the volume of the SPB solution in the tank is acceptable.

The addition of SR 3.1.7.3 requires verification of the SPB concentration above 8.0 percent by weight be performed every 31 days and once within 24 hours after water or boron is added to solution is needed to maintain the pH of the sump above 7.0 so that iodine re-evolution does not occur. This minimum requirement will not impact the reactivity control function of the SLC system because previously the minimum concentrations were controlled by SR 3.1.7.5, which specifies a minimum quantity of Boron-10 in the SLC solution tank and SR 3.1.7.6, which maintains the SLC conditions for meeting ATWS concerns. Neither of these SRs are being modified so the addition of a minimum SPB concentration to the TSs will not impact the reactivity control function of the SLC system and the SLC system will still meet GDC 26 and 29. Therefore, the addition of this new SR is acceptable.

These changes reflect the use of SLC for maintaining suppression pool pH following a LOCA involving fuel damage. As such, this change is already reflected in the analyses found acceptable above and is, therefore, acceptable with regard to DBA radiological consequences.

3.3.2 TS 3.6.4.1, Secondary Containment

TS 3.6.4.1, Secondary Containment, is being revised to delete the requirement for operability during core alterations and movement of irradiated fuel assemblies in the secondary containment. The effect of this change would be that the secondary containment could be inoperable during core alterations and movement of irradiated fuel assemblies.

The licensee states that the secondary containment is shared between all three units. Each units TSs require the secondary containment to be operable when any unit is in Mode 1 through 3. Thus, the only time that the secondary containment would be permitted to be inoperable is when all three units are in Modes 4 or higher.

In a letter dated July 2, 2004, the licensee stated as a defense-in-depth measure, TVA plans to revise the following procedures:

> The BFN General Operating Instruction for fuel movements during refueling will be revised to verify that prior to moving irradiated fuel, if secondary containment is not required to be operable, that it can be reestablished.
>
> The BFN Technical Instruction for secondary containment penetration breach analysis currently contains actions to be taken in the event secondary containment cannot be maintained due to a breach exceeding the available margin. These steps include stationing an Auxiliary Unit Operator at the breach location that is responsible for closing the breach if instructed by the control room. This instruction will be revised to require calculating the size of the breaches in secondary containment even when the TSs does not require secondary containment.
>
> The BFN Abnormal Operating Instruction for fuel damage during refueling provides the symptoms, automatic actions and operator actions for a fuel damage accident, including a dropped fuel bundle. Steps will be added to this instruction to ensure that secondary containment is intact or promptly restored following a postulated fuel handling accident.

The NRC staff concludes that these procedural revisions enhance the defense in depth by providing a defined set of procedures and actions to re-establish closure of the secondary containment in the event of a refueling accident. As such, the actions provide an increased measure of protection for public health and safety.

The NRC staff has reviewed the licensee's analysis which indicates that the dose guidelines of RG 1.183 would not be exceeded during a refueling accident. In addition, the licensee has indicated that administrative controls will be put into place to ensure closure of the secondary containment and terminate venting to support the design function of the secondary containment to serve as a barrier to the release of radioactive materials. Based on administrative controls being available to isolate the secondary containment in the event of a refueling DBA, the secondary containment being required operability while either of the other units are in Modes 1 through 3, the NRC staff finds the proposed change acceptable.

3.3.3 TS 3.6.4.2, Secondary Containment Isolation Valves

TS 3.6.4.2, Secondary Containment Isolation Valves (SCIVs) is being revised to delete the requirement for operability during core alterations and movement of irradiated fuel assemblies in the secondary containment. The affect of this change would be that the secondary containment isolation valves could be inoperable during core alterations and movement of irradiated fuel assemblies.

The licensee states that the secondary containment is shared between all three units. Each units TSs require the secondary containment isolation valves to be operable when any unit is in Mode 1 through 3. Thus, the only time that the secondary containment would be permitted to be inoperable is when all three units are in Modes 4 or higher.

The NRC staff has reviewed the licensee's analysis which indicates that the dose guidelines of Regulatory Guide 1.183 would not be exceeded during a refueling accident. In addition, the licensee has indicated that administrative controls will be put into place to assure closure of the secondary containment, including the secondary isolation valves and terminate venting to support the design function of the secondary containment to serve as a barrier to the release of radioactive materials.

Based on the secondary containment isolation valves remaining operable if any of the three units are in Modes 1 through 3 and that administrative controls will be available to isolate the secondary containment including the secondary containment isolation valves in the event of a refueling DBA, the NRC staff finds the proposed change acceptable.

3.3.4 TS 3.6.4.3, Standby Gas Treatment System

TS 3.6.4.3, Standby Gas Treatment System is being revised to delete the requirement for operability during core alterations and movement of irradiated fuel assemblies in the secondary containment. The SGTS draws a negative pressure on the secondary containment and exhaust through filters to the plant stack. The licensee states that the secondary containment is shared between all three units. Each unit has the requirement in the TSs that three subsystems of the SGTS will be operable when the respective unit is in Mode 1 through 3. Thus, the only time that the SGTS would be permitted to be inoperable is the condition in which all three units are in Modes 4 or higher.

The NRC staff has reviewed the licensee's analysis which indicates, that even if no credit is taken for the SGTS, the dose guidelines of RG 1.183 would not be exceeded during a refueling accident. In addition, the licensee has indicated that administrative controls will be put into place to assure closure of the secondary containment and terminate venting to support the design function of the secondary containment to serve as a barrier to the release of radioactive materials.

Based on the SGTS remaining operable if any of the three units are in Modes 1 through 3 and that administrative controls are available to isolate the secondary containment in the event of a refueling DBA, the NRC staff finds that the proposed change is acceptable.

3.3.5 TS 3.7.3, Control Room Emergency Ventilation System

TS 3.7.3 is being revised to delete the requirement for operability of CREVS during core alterations and movement of irradiated fuel assemblies in the secondary containment. The licensee states in his submittal that the three control rooms for the three units share the same control envelope. Each unit has the requirement in the TSs that the control room emergency ventilation system will be operable when the respective unit is in Mode 1 through 3. Thus, the only time that the control room emergency ventilation system would be permitted to be inoperable is the condition in which all three units are in Modes 4 or higher.

The NRC staff has reviewed the licensee's analysis which indicates that the dose guidelines of Regulatory Guide 1.183 would not be exceeded during a refueling accident. In addition, the licensee has indicated that administrative controls will be put into place to assure closure of the secondary containment and terminate venting to support the design function of the secondary containment to serve as a barrier to the release of radioactive materials and would reduce the potential source term to the common control room envelope.

Based on the CREVS instrumentation remaining operable if any of the three units are in Modes 1 through 3 and that administrative controls are available to isolate the secondary containment in the event of a refueling DBA. The use of administrative controls will ensure that adequate protection is afforded to the operators in the control rooms of all three units.

3.3.6 TS Table 3.3.6.2-1, Secondary Containment Isolation Instrumentation

TS Table 3.3.6.2-1, Secondary Containment Isolation Instrumentation, is being revised to delete footnote (b) which specifies the applicable modes or other specified conditions for operable secondary containment instrumentation during core alterations and movement of irradiated fuel assemblies in the secondary containment. The licensee states that the "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 does not take credit for the secondary containment function."

The effect of this change is that the secondary containment would not be automatically isolated on a high-radiation signal from the Reactor Zone Exhaust or Refueling Floor Exhaust. The Reactor zone exhaust is specific to each unit. Since the only DBA that could result in a high-radiation signal in a reactor zone during refueling is the FHA and since the isolation of the secondary containment is not required to be operable in the mitigation of a refueling DBA, a LCO based on 10 CFR 50.36 is not required and the NRC staff finds this change acceptable.

The Refueling Floor Exhaust high-radiation signal is common to all three units. If any of the three units is operating in Modes 1, 2, or 3, the technical specification for that unit will require secondary containment isolation instrumentation for that unit including the Refueling Floor Exhaust high-radiation signal to be operable. Thus, the proposed change which would allow the refueling Floor Exhaust high radiation instrumentation to be non-operable during refueling would only be valid when all three units are in Mode 4 or higher. With all three units in Mode 4 or higher, the only DBA that could result in a high-radiation signal is the FHA. The licensee has shown that the secondary containment is not required to be operable in the mitigation of a refueling DBA. Thus, a LCO based on 10 CFR 50.36 is not required and the NRC staff finds this change acceptable.

The NRC staff has reviewed the licensee's analysis which indicates that the dose guidelines of RG 1.183 would not be exceeded during a refueling accident. In addition, the licensee has indicated that administrative controls will be put into place to assure closure of the secondary containment and terminate venting to support the design function of the secondary containment stated in UFSAR Section 1.5.1.6 to serve as a barrier to the release of radioactive materials.

3.3.7 TS Table 3.3.7.1-1, Control Room Emergency Ventilation (CREV) System Instrumentation

Delete the requirement for operability during core alterations and movement of irradiated fuel in the secondary containment. The operability of these secondary containment and CREVS functions will no longer be assumed as an initial condition in the DBA FHA radiological consequence analyses. As such, Criterion 2 of 10 CFR 50.36 is no longer met. The NRC staff's review above of the DBA FHA analyses determined that the EAB, LPZ, and control room doses were acceptable without credit being taken for the above systems and functions

TS Table 3.3.7.1-1, Control Room Emergency Ventilation (CREV) System Instrumentation, is being revised to delete footnote (b) which specifies the applicable modes or other specified conditions for operable CREV instrumentation during core alterations and movement of irradiated fuel assemblies in the secondary containment.

The effect of deleting this footnote is that high-radiation signals from the Reactor Zone Exhaust, the Refueling Floor Exhaust, and the Control Room Air Supply Duct for the unit in refueling will not actuate the CREV system during core alterations and movement of irradiated fuel assemblies in the secondary containment. The Reactor zone exhaust is specific to each unit. Since the only DBA that could result in a high-radiation signal in a reactor zone during refueling is the FHA and since the CREV system is not required to be operable in the mitigation of a refueling DBA, an LCO based on 10 CFR 50.36 is not required and the NRC staff finds this change acceptable.

The Refueling Floor Exhaust high-radiation signal and the Control Room Air Supply Duct high-radiation signals are common to all three units. If any of the three units is operating in Modes 1, 2, or 3, the TS for that unit will require the CREVs instrumentation for that unit including the Refueling Floor Exhaust high-radiation signal and the Control Room Air Supply Duct Radiation signal to be operable. Thus, the proposed change which would allow the Refueling Floor Exhaust high-radiation instrumentation and the Control Room Air Supply Duct Radiation signal to be non-operable during refueling would only be valid when all three units are in Mode 4 or higher. With all three units in Mode 4 or higher, the only DBA that could result in a high-radiation signal is the FHA. The licensee has shown that the CREV system is not required to be operable in the mitigation of a refueling DBA. Thus, an LCO based on 10 CFR 50.36 is not required and the NRC staff finds this change acceptable.

The NRC staff has reviewed the licensee's analysis which indicates that the dose guidelines of RG 1.183 would not be exceeded during a refueling accident. In addition, the licensee has indicated that administrative controls will be put into place to assure closure of the secondary containment and terminate venting to support the design function of the secondary containment to serve as a barrier to the release of radioactive materials and would reduce the potential source term to the common control room envelope.

3.3.8 Addition of TS Section 3.9.9, Decay Time

The NRC staff has reviewed the inclusion of a decay-time specification as Section 3.9.9 in the TS. Section 10.5.5 of the UFSAR indicated that administrative controls are used to ensure that fuel pool heat load does not exceed available cooling capacity. The licensee has sized the capacity of the fuel pool cooling (FPC) and auxiliary decay heat removal Systems, considering seasonal cooling water temperatures and current heat exchanger conditions. These systems are utilized to maintain the fuel pool temperature at or below 125°F during normal refueling outages (average spent fuel batch discharged from the equilibrium fuel cycle). In addition, the RHR system can be operated in parallel with the FPC system (supplemental fuel pool cooling) to maintain the fuel pool temperature less than 150°F if a full core off load is performed.

The licensee has indicated that the decay time assumed in the fuel handling DBA is 24 hours. Inclusion of the decay time specification ensures that fuel will not be moved prior to the time assumed for the FHA to occur in the refueling DBA. This is consistent with the requirement of an LCO that in accordance with 10 CFR 50.36 Criterion 2, since the 24 hours is an input parameter used in the FHA. Based on the specification being consistent with the time and the format of the TS Section, have been determined to be acceptable.

3.3.9 Facility Operating License Condition 2.C.(4) for Units 2 and 3

The licensee proposed the deletion of Facility Operating License Condition 2.C.(4) for Units 2 and 3. This license condition required TVA to perform an analysis of the DBA LOCA to confirm compliance with GDC 19 and offsite limits considering MSIV and ECCS leakage. As required the licensee submitted the LOCA analysis in this amendment application. As discussed above, the NRC staff has determined that the licensee's radiological analyses for BFN Units 2 and 3 to be acceptable, and that TVA has met the action required by this license condition for Units 2 and 3. Therefore, the NRC staff finds the deletion of Facility Operating License Condition 2.C.(4) for Units 2 and 3 acceptable.

3.4 Environmental Qualification

While in development of RG 1.183, the NRC staff recognized that current environmental analysis may be impacted by modifications associated with AST implementation and decided that while the generic issue concerning the effect of increased cesium releases on equipment was being resolved, licensees are permitted to use either the AST or the TID-14833 assumptions for performing the required EQ analyses. The licensee stated the following:

> . . . TVA has elected to retain the TID-14844 assumptions for performing the required environmental qualification (EQ) analyses. The radiation doses used for the EQ 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. The BFN AST analysis considers the source term from the Fission Product Inventory shown on Table 2-2 of the Safety Assessment. A reactor thermal power of 4031 MWt (102 percent of 3952 MWt) is also used . . . . There are no instruments in the suppression pool that are within the EQ program . . . . The BFN EQ program is based on qualification for 100 days post accident. Based on the above, the continued use of the TID-14844 source term provides integrated doses for equipment which envelope [sic] those that would be calculated using AST. Therefore, following implementation of AST, BFN will continue to meet their commitment to 10 CFR 50.49 by using a radiation environment associated with the most severe design basis accident.

NUREG-1465, Accident Source Terms for Light-Water Nuclear Power Plants, discusses the findings that equipment dose calculations performed with this NUREG source term were lower than doses calculated with the TID-14844 source term during the gap release and early in-vessel release phases of core degradation. NUREG-0933 Issue 187, The Potential Impact of Postulated Cesium Concentration on Equipment Qualification, discussed the fact that analyses showed that for equipment exposed to the containment atmosphere, the TID-14844 source term, and the gap and in-vessel releases in the AST produced similar integrated doses. The NUREG also indicated that for equipment exposed to suppression pool water, the integrated doses calculated with the AST remain enveloped by those calculated with TID-14844 for the first 145 days post accident for a BWR, including the 30 percent vs. one percent release of cesium. It was concluded in this NUREG that there was no clear basis for backfitting the requirement to modify the design basis for equipment qualification to adopt the AST. There would be no discernible risk reduction associated with such a requirement.

The NRC staff reviewed the licensee's submittal regarding EQ at the proposed reactor thermal power of 4031 MWt against the assumptions in RG 1.183, NUREG-1465, and NUREG-0933. For the equipment exposed to sump (suppression pool) water, the integrated doses calculated with the AST exceeded those calculated with TID-14844 after 145 days for a BWR, because of the 30 percent vs. 1 percent release of cesium according to NUREG-1465. However, as there is no EQ equipment in the sump and the licensee has opted for continued use of TID-14844 source term which provides integrated doses for equipment enveloping those that would be calculated using AST as the current licensing basis for a post accident period of 100 days, the NRC staff finds the dose impacts at a reactor thermal power of 4031 MWt (102 percent of 3952 MWt) acceptable.

## 3.5 Summary

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the BFN Unit 1, 2, and 3 design bases is superseded by the AST proposed by TVA. The previous offsite and control room accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR 50.67 or small fractions thereof, as defined in RG 1.183. All future radiological analyses performed to demonstrate compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as described in the now-updated BFN Unit 1, 2, and 3 design bases.

Based on the above information, the NRC staff considered the applicability of the revised source terms to operating reactors and determined that the current analytical approach based on the TID-14844 source term would continue to be adequate to protect public health and safety, and that operating reactors licensed under this approach would not be required to reanalyze accidents using the revised source term. The NRC staff compared the doses estimated by TVA to the applicable regulatory criteria and finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and control room doses will continue to comply with these criteria.

As discussed above, to support the implementation, the NRC staff has concluded that although not designated safety related, the SLCS as installed at BFN is a system of sufficient quality that provides reasonable assurance that the SPB will be injected into the core upon activation. However to ensure the reliability, redundancy, and quality of the system is maintained the following condition is being added to the license:

> The licensee shall maintain the Augmented Quality Program for the Standby Liquid Control system to provide quality control elements to ensure component reliability for the required alternative source term function defined in the Updated Final Safety Analyses Report.

Since these analyses were performed at a power level of 4031 MWt (102 percent of 3952 MWt), the NRC staff finds that the radiological consequences of these DBAs would remain bounding up to a rated thermal power of 3952 MWt. However, the approval of this amendment does not constitute authority to operate above the current licensed rated thermal power.

Additionally, the NRC staff reviewed the seismic ruggedness of the structures and components associated with the MSIV alternative drain path for Unit 1. The approval of this amendment does not constitute a change in the licensing basis of the alternative drain path for Unit 1, it does represent the NRC staff's acceptance of the alternative drain path methodology proposed for use to implement AST. As described above, TVA must receive approval of the change in licensing and design bases as well as complete actions necessary to confirm, by means of a seismic analysis, the conclusions obtained via the use of approximate calculations for the Browns Ferry Unit 1 turbine building prior to the restart of the Unit 1. This requirement concerning the Unit 1 turbine is contained in the following license condition:

> The licensee is required to confirm that the conclusions made in TVA's letter dated September 17, 2004, for the turbine building remain acceptable using seismic demand accelerations based on dynamic seismic analysis prior to the restart of Unit 1.

## 4.0 STATE CONSULTATION

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

## 5.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (67 FR 636977 and 69 FR 22883). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

## 6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the

Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Steven LaVie, Undine Shoop, Nitin Patel, James Bongarra, Krzysztof Parczewski, Kahtan Jabbour, Edwin Forrest, Thomas Koshy, John Ma, C. Craig Harbuck, Devender Reddy

Attachments: 1. Table 1
2. Table 2

Date: September 27, 2004

Table 1

BFN Accident Analysis Parameters

**All Accidents**

| | |
|---|---|
| Reactor power (3952 x 1.02), MWt | 4031 |
| Core peaking factor | 1.5 |
| Main condenser volume, $ft^3$ | 122,400 |
| SGTS Flow, cfm | |
| Stack, Elevated | 24750 |
| Damper bypass, ground level | 10 |
| SGTS drawdown time, min | <2 |
| SGTS HEPA filter efficiency, particulate, % | 90 |
| SGTS Charcoal Filter Efficiency, % | Not credited |
| Dose conversion factors | FGR11/FGR12 |
| Breathing rate, offsite, $m^3/s$ | |
| 0-8 hours | 3.5E-4 |
| 8-24 hours | 1.8E-4 |
| >24 hours | 2.3E-4 |
| Breathing rate, control room,$m^3/s$ | 3.5E-4 |
| Control room normal intake flow, cfm | 6717 |
| Control room unfiltered infiltration, cfm | 3717 |
| Control room filtered pressurization, cfm | 3000 |
| Control room volume, $ft^3$ | 210,000 |
| Control room intake HEPA filter efficiency, particulate, % | 90 |
| Control room charcoal filter efficiency, % | Not credited |
| Control room occupancy factor | |
| 0-24 hrs | 1.0 |
| 1-4 days | 0.6 |
| 4-30 days | 0.4 |

**Control Rod Drop Accident (RDA)**

| | |
|---|---|
| Fraction of core Inventory in gap | |
| Noble gases | 0.1 |
| Iodine | 0.1 |
| Br | 0.05 |
| Cs,Rb | 0.12 |
| Failed rods | 850 |
| Fraction of failed rods that reach melt | 0.0077 |

| | |
|---|---|
| Melted fuel release fraction to vessel | |
| Noble gases | 1.0 |
| Iodine | 0.5 |
| Br | 0.3 |
| Cs,Rb | 0.25 |
| Te | 0.05 |
| Ba, Sr | 0.02 |
| Noble metals | 0.0025 |
| Ce | 0.0005 |
| La | 0.0002 |
| Fraction of activity released to vessel that enters main condenser | |
| Noble gases | 1.0 |
| Iodine | 0.1 |
| others | 0.01 |
| Fraction of activity released from main condenser | |
| Noble gases | 1.0 |
| Iodine | 0.1 |
| others | 0.01 |
| Main condenser (plus LP turbine) free volume, $ft^3$ | 187,000 |
| Release rate from main condenser, cfm | 1850 |
| Release duration, days | 30 |

## **Loss of Coolant Accident**

Containment Leakage Source

Onset of gap release phase, min 2.0

Core release fractions and timing–CNMT atmosphere

| Duration, hrs | 0.5000E+00 | 0.1500E+01 |
|---|---|---|
| Noble Gases: | 0.5000E-01 | 0.9500E+00 |
| Iodine: | 0.5000E-01 | 0.2500E+00 |
| Cesium: | 0.5000E-01 | 0.2000E+00 |
| Tellurium: | 0.0000E+00 | 0.5000E-01 |
| Strontium: | 0.0000E+00 | 0.2000E-01 |
| Barium: | 0.0000E+00 | 0.2000E-01 |
| Noble Metals: | 0.0000E+00 | 0.2500E-02 |
| Cerium: | 0.0000E+00 | 0.5000E-03 |
| Lanthanum: | 0.0000E+00 | 0.2000E-03 |

| | |
|---|---|
| Iodine species distribution | |
| Elemental | 0.95 |
| Organic | 0.0485 |
| Particulate | 0.0015 |

| | |
|---|---|
| Primary CNMT volume, $ft^3$ | |
| Drywell | 159,000 |
| Supression pool air space | 119,400 |
| CNMT leakrate, %/day | 2.0 |
| Unit 1 Secondary containment volume (50% of free volume) | 1,311,209 |
| Unit 2 and 3 Secondary containment volume (50% of free volume) | 1,931,500 |
| Hardened wet well vent release (elevated), 8 hours to 30 days, scfh | 10 |
| Release via SGTS (elevated) and base of stack (ground) | |
| SGTS ground level leakage (base of stack), cfm | 10 |
| Volume at base of stack (50% of free volume), $ft^3$ | 34,560 |
| Drywell natural deposition | |
| Particulate | Powers 10%-percentile Model |
| Elemental | Same as particulate |
| Surface area for elemental iodine deposition in drywell, $m^2$ | 3409 |
| Drywell maximum accident conditions | |
| Pressure, psig | 48.3 |
| Temperature, degF | 294.9 |
| Control room isolation delay, minutes | 10 |

<u>CAD System Release</u>

| | |
|---|---|
| Activity same as CNMT leakage case | |
| Flow rate, cfm | 139 |
| CAD operation, days post accident | 10, 20, 29 |
| CAD operation duration, hours | 24 |
| No mixing in RB, release via elevated release point | |

<u>MSIV Leakage</u>*

| | |
|---|---|
| Activity same as CNMT leakage case | |
| MSIV TS leak rate @25 psig, scfh | |
| One line | 100 |
| Total | 150 |

Main steam line configuration for deposition analysis

all four steam lines intact, in service at start of event
One inboard MSIV fails to close
In each of three isolated lines, a well-mixed control volume exists
Only horizontal lines are credited
100 scfh is assumed to exist in faulted line
One of remaining lines is assumed to leak at 50 scfh; other two are leaktight
Pressure between closed MSIVs is assume to be equal to CNMT pressure
Temperature is assumed to be normal steam line conditions
Pressure downstream of outboard MSIVs (and inboard MSIV on faulted line) is assumed to be atmospheric; normal operating temperature

MSIV leakage at test pressure is converted to volumetric flow based on post-LOCA drywell temperature and pressure
RADTRAD Bixler model used for elemental iodine
MSIV leakage from condenser is released without dilution or holdup in turbine building

| | | |
|---|---|---|
| MSIV Leakage that bypasses main condenser,% of total | | 0.5 |
| Steam line deposition | Aerosol | Elemental |
| Steam line | 99.87 | 99.01 |
| MC bypass | 89.33 | 16.37 |

<u>ECCS Leakage</u>

| | |
|---|---|
| Iodine species fraction | <u>Sump</u> |
| Particulate/aerosol | 0 |
| Elemental | 97 |
| Organic | 3 |
| Suppression pool liquid volume, $ft^3$ | 121,500 |
| Estimated leakage, gpm | 5 |
| Iodine Flash Fraction | 0.1 |
| SGTS charcoal filtration | Not credited |

Release via SGTS (elevated) and base of stack (ground)

**<u>Fuel Handling Accident</u>**

| | |
|---|---|
| Fuel rods damaged *(conservatively based on 7 x 7 fuel)* | 111 |
| Decay period, hrs | 24 |
| Fraction of core in gap | |
| I-131 | 0.08 |
| Kr-85 | 0.1 |
| Other iodines | 0.05 |
| Other noble gases | 0.05 |
| Pool decontamination factor | 200 |
| Release period | Instantaneously |
| Hold-up and mitigation | No credit |
| Release via: | RB refueling zone vent |

## Main Steam Line Break

| | |
|---|---|
| Reactor coolant activity, uCi/gm dose equivalent I-131 | |
| Normal | 3.2 |
| Spike | 32 |
| Mass release, lbm | |
| Steam | 11,975 |
| Liquid (saturated at 898 psia) | 42,215 |
| Release flash fraction (pressure=1020 psia) | 0.38 |
| Release duration, sec | 5.5 |
| Iodine species | 100% Elemental |

Table 2

BFN METEOROLOGY

| Time Period | Control Room Unit 1 | Control Room Unit 3 | Site Boundary EAB | LPZ |
|---|---|---|---|---|
| Top of Stack Releases (LOCA and CRDA) | | | | |
| 0-0.5 hrs | 3.40E-5 | * | 2.35E-5 | 1.26E-5 |
| 0.5-2 hrs | ** | 1.41E-7 | 1.19E-6 | 1.13E-6 |
| 2-8 hrs | ** | 4.50E-8 | | 5.75E-7 |
| 8-24 hrs | ** | 2.54E-8 | | 4.10E-7 |
| 1-4 days | ** | 7.36E-9 | | 1.97E-7 |
| 4-30 days | ** | 1.24E-9 | | 6.88E-8 |
| Base of Stack Releases (LOCA and CRDA) | | | | |
| 0-2 hrs | 2.00E-4 | * | 2.62E-4 | 1.31E-4 |
| 2-8 hrs | 1.28E-4 | * | | 6.61E-5 |
| 8-24 hrs | 5.72E-5 | * | | 4.69E-5 |
| 1-4 days | 4.05E-5 | * | | 2.23E-5 |
| 4-30 days | 3.09E-5 | * | | 7.96E-6 |
| Turbine Building Exhaust Releases (MSLB Only, Unit 1 Post LOCA MSIV Leakage) | | | | |
| 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 |
| Refuel Floor Releases | | | | |
| 0-2 hrs | 4.60E-4 | * | 2.62E-4 | 1.31E-4 |
| Turbine Building Roof Ventilators (Units 2 and 3 Post-LOCA MSIV Leakage) | | | | |
| 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 |

* Unit 1 intake limiting
** Unit 3 intake limiting

Due to dual intake configuration, limiting $\chi/Q$ needs to be divided by two.