Enclosure 1

Edwin I. Hatch Nuclear Plant
Request to Implement an Alternative Source Term

AST Safety Assessment

## TABLE OF CONTENTS

## LIST OF TABLES

## LIST OF FIGURES

## ACRONYMS AND ABBREVIATIONS

| | |
|---|---|
| $\Delta h$ | Head Loss |
| $\lambda$ | Removal Coefficient |
| $\rho$ | Density |
| $\chi/Q$ | Atmospheric Dispersion Factor |
| ALT | Alternate Leakage Treatment |
| AST | Alternative Source Term |
| ASTM | American Society for Testing and Materials |
| Btu | British Thermal Unit |
| BWR | Boiling Water Reactor |
| BWROG | Boiling Water Reactor Owners' Group |
| CAD | Containment Atmosphere Dilution |
| CEDE | Committed Effective Dose Equivalent |
| cfh | Cubic Feet per Hour |
| cfm | Cubic Feet per Minute |
| CFR | Code of Federal Regulations |
| Ci | Curie |
| cm | Centimeter |
| $C_p$ | Pressure Coefficient |
| CRDA | Control Rod Drop Accident |
| CS | Core Spray |
| CSB | Containment Systems Branch |
| CsI | Cesium Iodide |
| CST | Condensate Storage Tank |
| DBA | Design Basis Accident |
| DBE | Design Basis Earthquake |
| DDE | Deep Dose Equivalent |
| DE I-131 | Dose Equivalent I-131 |
| DF | Decontamination Factor |
| DNBR | Departure from Nucleate Boiling Ratio |
| DW | Drywell |
| EAB | Exclusion Area Boundary |

## ACRONYMS AND ABBREVIATIONS

| | |
|---|---|
| ECCS | Emergency Core Cooling System |
| EDE | Effective Dose Equivalent |
| EPRI | Electric Power Research Institute |
| ESF | Engineered Safety Feature |
| F | Fahrenheit |
| FGR | Federal Guidance Report |
| FHA | Fuel Handling Accident |
| FSAR | Final Safety Analysis Report |
| ft | Feet |
| g | Gram |
| GDC | General Design Criterion |
| GIP | Generic Implementation Procedure |
| GL | Generic Letter |
| gpm | Gallons per Minute |
| HNP | Edwin I. Hatch Nuclear Plant |
| HPCI | High-Pressure Coolant Injection |
| HPT | High Pressure Turbine |
| hr | Hour |
| HVAC | Heating, Ventilation, and Air Conditioning |
| ICRP | International Commission on Radiological Protection |
| ILRT | Integrated Leak Rate Test |
| in | Inch |
| INEEL | Idaho National Engineering and Environmental Laboratory |
| kg | Kilogram |
| KI | Potassium Iodide |
| kV | Kilovolt |
| lbm | Pounds (Mass) |
| LOCA | Loss-of-Coolant Accident |
| LOSP | Loss of Offsite Power |
| LPZ | Low Population Zone |
| m | Meter |
| MCC | Motor Control Center |

## ACRONYMS AND ABBREVIATIONS

| | |
|---|---|
| MCR | Main Control Room |
| MCREC | Main Control Room Environmental Control |
| MeV | Million Electron Volts |
| min | Minute |
| mm | Millimeter |
| MOV | Motor Operated Valve |
| mph | Miles per Hour |
| MSIV | Main Steam Isolation Valve |
| MSL | Main Steam Line |
| MSLB | Main Steam Line Break |
| MWt | Megawatt Thermal |
| NRC | Nuclear Regulatory Commission |
| Pa | Pascal |
| psia | Pounds Per Square Inch Absolute |
| psig | Pounds Per Square Inch Gauge |
| PSW | Plant Service Water |
| PWR | Pressurized Water Reactor |
| Q | Volumetric Flow Rate |
| RB | Reactor Building |
| RBCCW | Reactor Building Closed Cooling Water |
| RCIC | Reactor Core Isolation Cooling |
| RCS | Reactor Coolant System |
| rem | Roentgen Equivalent Man |
| RG | Regulatory Guide |
| RHR | Residual Heat Removal |
| RWCU | Reactor Water Cleanup |
| scfh | Standard Cubic Feet per Hour |
| sec | Second |
| SER | Safety Evaluation Report |
| SGTS | Standby Gas Treatment System |
| SLC | Standby Liquid Control |
| SNC | Southern Nuclear Operating Company |

## ACRONYMS AND ABBREVIATIONS

| | |
|---|---|
| SPB | Sodium Pentaborate ($Na_2O$-$5B_2O_3$-$10H_2O$) |
| SQUG | Seismic Qualification Utility Group |
| SRP | Standard Review Plan |
| SSE | Safe Shutdown Earthquake |
| TB | Turbine Building |
| TEDE | Total Effective Dose Equivalent |
| TID | Technical Information Document |
| TIP | Traversing Incore Probe |
| TS | Technical Specification |
| TSC | Technical Support Center |
| USI | Unresolved Safety Issue |
| wg | Water Gauge |

# 1. INTRODUCTION

## 1.1 Overview and Objectives

The objective of this safety assessment is to document implementation of the Alternative Source Term (AST) for the Edwin I. Hatch Nuclear Plant (HNP) Units 1 and 2. The implementation of AST is governed by Title 10 of the Code of Federal Regulations (CFR), Section 50.67, the guidelines of the Standard Review Plan (SRP) Section 15.0.1, and Regulatory Guide (RG) 1.183.

Conformance to the positions of RG 1.183 is closely adhered to for AST implementation. A RG 1.183 conformance matrix is included as Appendix A to this enclosure, providing the RG 1.183 positions, the corresponding HNP positions, and any clarifying comments. In addition, due consideration has been given to U.S. Nuclear Regulatory Commission (NRC) Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms," March 7, 2006.

Southern Nuclear Operating Company (SNC) has elected to perform a full scope implementation of the AST for HNP Units 1 and 2 as defined in RG 1.183. The implementation consists of the following:

1. Identification of the core source term based on plant specific analysis of core fission product inventory. All characteristics of the AST are considered, including the composition and magnitude of the radioactive material, its chemical and physical form, and the timing of its release.
2. Determination of the release fractions for the four Final Safety Analysis Report (FSAR) Boiling Water Reactor (BWR) Design Basis Accidents (DBAs) that could potentially result in significant control room and offsite doses. These DBAs are the loss-of-coolant accident (LOCA), the fuel handling accident (FHA), the control rod drop accident (CRDA), and the main steam line break (MSLB).
3. Calculation of fission product deposition rates and removal efficiencies.
4. Calculation of offsite, control room, and technical support center (TSC) personnel total effective dose equivalent (TEDE).
5. Evaluation of suppression pool pH to ensure that the particulate iodine deposited into the suppression pool during a DBA LOCA does not re-evolve and become airborne as elemental iodine.
6. Evaluation of other related design and licensing bases such as NUREG-0737 requirements.

## 1.2 Summary of Technical Specification Changes

The implementation of AST and the radiological dose consequence analyses includes several changes to HNP Technical Specifications (TSs). Table 1 identifies the TS changes proposed. For a more detailed description and justification of these TS changes, see Enclosure 2 of this submittal.

**Table 1. Proposed Technical Specification Changes**

| TS Number | Scope and Description of Technical Specification Change | |
|---|---|---|
| 1.1 | Definition of DE I-131 | The definition of DE I-131 is revised to replace "thyroid dose" with "Committed Effective Dose Equivalent" and to reference FGR 11 for the dose conversion factors used in calculating I-131 concentration. |
| 3.4.6 | Reactor Coolant System Specific Activity | The maximum allowed reactor coolant specific activity is reduced from 4.0 μCi/g DE I-131 to 2.0 μCi/g DE I-131. |
| 3.6.1.3 | Secondary Containment Bypass Leakage | The maximum allowed bypass leakage rate for all secondary containment bypass leakage paths is 2.0% of the maximum allowable primary containment leakage rate. This is a new TS for Unit 1 and an increase from 0.9% of the maximum allowable primary containment leakage rate for Unit 2. |
| 3.6.1.3 | MSIV Leakage | The maximum allowable combined MSIV leakage rates are revised by increasing Unit 1 and decreasing Unit 2 to 100 scfh and eliminating the per line leakage limit. In addition, two separate surveillance acceptance criteria will be provided dependent on leakage rate test pressure. |
| 3.6.2.5 | DW Spray | A new TS for RHR DW spray is added to reflect the crediting of DW spray as part of the AST LOCA assumptions. |

## 1.3 Changes to Main Control Room Unfiltered Inleakage Assumptions

The HNP current licensing basis main control room (MCR) unfiltered inleakage limit is 110 cfm based on the administration of potassium iodide (KI) tablets to MCR occupants within 2 hr after the start of a design basis LOCA. The HNP Units 1 and 2 common MCR has a unique location. The MCR, as part of the control building, is located between the open end bays of the HNP Units 1 and 2 turbine buildings (TBs). The majority of the ductwork associated with the main control room environmental control (MCREC) system, which encompasses two independent filter trains for pressurizing the control room post-accident, is located external to the control room boundary on top of the control building within the confines of the HNP Units 1 and 2 TBs.

By letters dated August 4, 2003, March 29, 2004, October 27, 2004, and November 10, 2005, SNC submitted a course of action for developing responses to NRC Generic Letter (GL) 2003-01, "Control Room Habitability" information requests for HNP. GL 2003-01 was written to inform licensees that the design basis assumptions used for control room unfiltered inleakage, even with a pressurized control room, could be non-conservative. This was validated through testing at several power reactor facilities using the standard test method described in American Society for Testing and Materials (ASTM) consensus standard E741, "Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution."

In order to address the possibility of unfiltered inleakage into the HNP control room, the incorporation of KI was approved on an interim basis as a measure to limit the thyroid dose to control room occupants in the event of a design basis LOCA. The incorporation of KI in the interim licensing basis is provided to assure that the 30-day thyroid dose remains within the regulatory limits of 10 CFR 50, Appendix A, General Design Criterion (GDC) 19, with MCR unfiltered inleakage up to 110 cfm. As a condition of the licensing basis, the crediting of KI in limiting post-LOCA doses to MCR personnel is for an interim period, expiring on May 31, 2010.

Tracer gas testing of the MCR envelope was completed in June 2006 using ASTM consensus standard E741. The most limiting results from testing revealed 5 cfm unfiltered inleakage into the MCR. With the completion of tracer gas testing, SNC will be completing its response to GL 2003-01 under a separate letter.

The change to the licensing basis by the implementation of AST is required to comply with control room habitability regulatory requirements without relying on the KI interim licensing basis. Approval of the AST license amendment request will ensure that the design basis radiological analysis for occupants of the MCR reflects the most limiting unfiltered inleakage into the MCR. There is significant margin between the measured unfiltered inleakage and the unfiltered inleakage assumptions used in the most limiting DBA dose consequence analysis for occupants of the MCR, which is the LOCA. Assumed unfiltered inleakage for the LOCA in the AST dose consequence analysis is 115 cfm.

## 1.4 Summary

Implementation of the AST as the HNP radiological consequence analyses licensing basis requires a license amendment in accordance with the requirements of 10 CFR 50.67. The AST radiological consequence analyses demonstrate that the offsite, MCR, and TSC post-accident radiological doses remain within regulatory limits.

# 2. EVALUATION

## 2.1 Changes to Current Licensing Basis

Implementation of AST includes several changes to the current licensing basis. These are summarized below.

### 2.1.1 MSIV Alternate Leakage Treatment

RG 1.183, Appendix A, Section 6, allows credit for a reduction in main steam isolation valve (MSIV) releases due to holdup and retention in the main steam line (MSL) piping downstream of the MSIV and in the condenser for a DBA LOCA. This credit is based, in part, on the piping and components of the alternate leakage treatment (ALT) release path being capable of performing their safety functions during and after a design basis earthquake (DBE). The HNP AST implementation credits the ALT pathway for HNP Unit 1.[1] Section 2.7.1 describes the ALT application.

### 2.1.2 MCR and TSC Inleakage

The HNP current licensing basis contains an MCR unfiltered inleakage limit of 110 cfm based upon the administration of KI. Implementation of the AST would allow the interim licensing basis crediting KI to be retired while also increasing the MCR design basis unfiltered inleakage limit to 115 cfm. TSC inleakage is also considered in evaluating the dose consequences to occupants of the TSC.

### 2.1.3 Standby Liquid Control System

The standby liquid control (SLC) system is credited for the injection of sufficient sodium pentaborate (SPB) solution to prevent the re-evolution of iodine from the suppression pool for a 30-day period

[1] NRC has previously approved MSIV ALT for Unit 2 (Reference 1).

following a DBA LOCA. The pH buffering effect of SLC injection is sufficient to offset the effects of acids that are transported to the suppression pool and maintain suppression pool pH at or above 7, thus precluding the re-evolution of elemental iodine. NRC review guidelines, "Guidance on the Assessment of a BWR SLC System for pH Control," were addressed. An evaluation of the SLC system and its ability to perform the post-LOCA injection function is discussed in Section 2.7.2.

### 2.1.4 Turbine Building Ventilation

The TB ventilation system is credited for the removal of activity from the TB beginning 9 hr after the start of a DBA, exhausting at a rate of 15,000 cfm. TB ventilation is credited for the LOCA, CRDA, and MSLB.

### 2.1.5 Secondary Containment Bypass Leakage

The primary containment leakage that bypasses the secondary containment (reactor building) is assumed to be into the condenser for evaluating MCR doses for the DBA LOCA analysis. Activity holdup and deposition in the condenser from this secondary containment bypass leakage is credited in a manner similar to the treatment of MSIV releases and the MSIV ALT pathway.

An evaluation of the Unit 1 secondary containment system was performed using the guidance provided by Branch Technical Position CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants." All primary containment penetrations were assessed to identify the leakage paths that do not terminate within the secondary containment and should be considered as potential secondary containment bypass leakage paths. Table 2 lists the piping systems identified as potential bypass leakage paths. This evaluation was performed consistent with the current licensing basis for Unit 2.

### 2.1.6 Drywell Sprays

Drywell (DW) sprays are credited to help remove airborne particulates in the DW in the case of the DBA LOCA. DW sprays are also credited in the DBA LOCA analysis for primary containment atmosphere temperature and pressure reduction. This temperature and pressure reduction over time allows primary containment leakage and MSIV leakage to be reduced by 50% at 72 hr after the initiation of the LOCA. The primary containment pressure and temperature profiles over time for the DBA LOCA were developed by GE, consistent with the current licensing basis containment analysis of the DBA LOCA.

Manual activation of sprays is required by control room operators, and is assumed to be manually initiated following a DBA LOCA (beginning of piping break). Initiation is based on radiation levels in the DW.

### 2.1.7 Atmospheric Dispersion Factors

The onsite atmospheric dispersion factors ($\chi/Q$ analysis) used for the radiological dose consequence analyses for both the MCR and the TSC are re-calculated for AST implementation. The current analysis is based on one year of meteorological data and the ARCON95 code. For AST, the analysis is based on a set of 3-year meteorological data and is performed with the ARCON96 code. No changes are made to the offsite atmospheric dispersion factors.

### 2.1.8 Cable Spreading Room Fans

For HNP, the limits of unfiltered inleakage credited in the dose estimates to occupants of the MCR takes into account the operation of the MCREC system in pressurization mode. Currently, the cable spreading room supply and exhaust fans, 1Z41-C009 and 1Z41-C010, are secured via operator action when the control room is pressurized to preclude a potential malfunction of those fans which could impact the capability to maintain the control room at a positive pressure.

A modification of the fan logic is planned to provide automatic securing of the cable spreading room supply and exhaust fans on automatic initiation of the pressurization mode of the MCREC system. This non-outage modification is scheduled to be completed by December 31, 2007.

### 2.1.9 FHA Decontamination Factor

For the FHA, a new decontamination factor (DF) for iodine in the spent fuel pool is determined. The two regions considered for the FHA are the area over the reactor core and the spent fuel pool. The minimum depth of water over the core is 23 ft. The minimum depth of water over the fuel in the spent fuel pool is 21 ft. The iodine DF derived in RG 1.183 assumes a water depth of 23 ft. Because the depth of water over the fuel in the spent fuel pool is less than 23 ft, a DF consistent with 21 ft of water is determined for use in the FHA dose analysis in Section 2.5.3.

### 2.1.10 Dose Equivalent I-131

The definition of dose equivalent I-131 (DE I-131) is revised to support AST implementation. "Thyroid dose" in the current definition is replaced with "committed effective dose equivalent" to more accurately depict the applicable dose component. Additionally, only Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988, is referenced for the dose conversion factors used in calculating DE I-131.

The current limit for DE I-131 specific activity in the primary coolant is 4.0 μCi/g. Analysis of the DBA MSLB uses a design input maximum DE I-131 concentration of 2.0 μCi/g. A TS change to reduce the maximum DE I-131 concentration from 4.0 μCi/g to 2.0 μCi/g is part of the AST implementation. The MSLB analysis calculates doses to occupants of the MCR based upon a DE I-131 concentration of 2.0 μCi/g.

**Table 2. Unit 1 Potential Secondary Containment Bypass Leakage**

| System Name | Pipe Service Description | Containment Isolation Valve Size (in) | Line Size[(1)] (in) | Line Quantity[(2)] | Remarks |
|---|---|---|---|---|---|
| Nuclear Boiler System | Main Steam to main turbine | 24 (each) | 24 | 4 | Design bases MSIV leakage has been considered in the dose consequence analyses following an accident. |
| | Condensate Drain | 3 | 3 | 1 | |

| System Name | Pipe Service Description | Containment Isolation Valve Size (in) | Line Size[1] (in) | Line Quantity[2] | Remarks |
|---|---|---|---|---|---|
| | Reactor Feedwater Supply | 18 (each) | 18 | 2 | Leakage through these lines must flow through three 18-in check valves in series per line before release to the TB. |
| CS System | Pump Condensate Supply for Test | 16 | 14 | 1 | Note 3 |
| HPCI System | Steam supply to the HPCI turbine | 10 | 1 | 1 | |
| | Pump Condensate Suction | 16 | 16 | 1 | Note 3 |
| | Pump Flow Test Line | 18 | 10 | 1 | Leakage must pass through a normally closed MOV that directs flow to the CST via the test line. |
| RCIC System | Steam Supply to the RCIC Turbine | 4 | 1 | 1 | |
| | Pump Condensate Suction | 6 | 6 | 1 | Note 3 |
| | Pump Flow Test Line | 18 | 10 | 1 | Leakage must pass through a normally closed MOV that directs flow to the CST via the test line. |
| RWCU System | RWCU Drainage to the main Condenser | 6 | 4 | 1 | Note 5 |
| | Drainage to Radwaste | 6 | 4 | 1 | Note 5 |
| Torus Drainage and Purification System | Torus Drainage to Condenser or Suction to Condensate Pumps | 8 | 3 | 1 | Normally isolated with a minimum of 3 normally closed valves. Also, Note 6. |
| | Torus Drainage to the Condensate Booster Pumps | 8 | 6 | 1 | Normally isolated with a minimum of 3 normally closed valves. Also, Note 6. |
| | Torus Drainage to the waste Surge Tank | 8 | 4 | 1 | Normally isolated with 3 manual normally closed valves. Also, Note 6. |

| System Name | Pipe Service Description | Containment Isolation Valve Size (in) | Line Size[(1)] (in) | Line Quantity[(2)] | Remarks |
|---|---|---|---|---|---|
| Radwaste | DW Equipment Drain Sump Discharge | 3 | 3 | 1 | Note 4 |
| | DW Floor Drain Sump Discharge | 3 | 3 | 1 | Note 4 |
| DW Pneumatic System | Nitrogen Supply from Nitrogen Storage Tanks to DW Pneumatic System | 2 (each) | 2 | 1 | Note 7 |
| | DW Pneumatic Suction | 1 | N/A | N/A | Suction path to the DW pneumatic system compressor has been permanently capped inside the RB. The drain line drains into the RB equipment drain sump and does not bypass the secondary containment. |
| Neutron Monitoring System | TIP Nitrogen Purge Supply | 3/8 | 2 | 1 | Note 9 |
| Primary Containment Purge and Inerting System | DW purge Supply and Nitrogen Make-up | 18, 6, 2 | 2, 6 | 2 | Notes 7, 8 |
| | DW Exhaust | 18, 2, 2, 2 | 18 | 1 | Processed by the SGTS. |
| | Torus Purge Supply and Torus Nitrogen Make-up | 18, 6, 2 | 2, 6 | 2 | Note 7, 8 |
| | Torus Exhaust | 18, 2, 2, 2 | 18 | 1 | Processed by the SGTS. |
| | Vacuum Breaker Air Supply | 1/2 | 4 | 1 | The path is normally isolated by 1T48-F342A-L (normally closed). The line will be under instrument air pressure (higher than the torus pressure) if valve 1T48-F342A-L is open. |
| Plant Service Water System | PSW Supply to DW Coolers | 8 | 10 | 1 | Closed loop system inside primary containment. |

| System Name | Pipe Service Description | Containment Isolation Valve Size (in) | Line Size[1] (in) | Line Quantity[2] | Remarks |
|---|---|---|---|---|---|
| | PSW Return from DW Coolers | 8 | 30 | 1 | Closed loop system inside primary containment. |
| RBCCW System | RBCCW Supply | 4 | 14 | 1 | Closed loop system inside primary containment. |
| | RBCCW Return | 4 | 14 | 1 | Closed loop system inside primary containment. |
| Demineralized Water System | Demineralized Water Supply to the Hose Stations Inside the DW | 1-1/2 | 4 | 1 | Isolation valve located outside the DW is locked closed during normal plant operation. |
| Primary Containment Integrated Leak Test System | ILRT Sample Line | 3/4 | 3/4 | 1 | A minimum of 3 isolation valves located outside the DW are closed during normal plant operation. |

Notes:
1. Pipe size at the secondary containment wall.
2. Total number of lines that pass through the secondary containment.
3. The lines for HPCI and RCIC system pump suction piping, the CS system, CST suction piping, and the torus drainage and purification influent piping from the CST are continuously filled with water from the CST to the isolation valve, and with suppression pool water to the pump side of the isolation valve. Therefore, no leakage to the environment is postulated to occur.
4. The containment drainage sumps are located in the base of the DW and are flooded with coolant following the postulated LOCA. This flooding creates a water seal inside the containment up to the closed isolation valves. These valves are leak tested in accordance with 10 CFR 50, Appendix J, and their leakage rates form a part of the total bypass leakage fraction.
5. The RWCU system is isolated from the nuclear process through the closure of two 6-in. isolation valves in series on the influent line, and through the closure of the 18-in. feedwater system check valves at the system effluent, as well as a 3-in. RWCU system effluent check valve. The path to the RCIC is normally isolated. The leakage estimated is the combined leakage rate through the 6-in. isolation valves and the 18-in. feedwater check valves. Directing drainage to either the radwaste system or the main condenser does not affect the estimate of bypass leakage since both 4-in. lines connect to the RWCU system loop via the 6-in. and 18-in. isolation valves. See drawing nos. H16062, H16063, H16145, H16188, H16189.
6. The effluent torus drainage and purification system line, by virtue of its location with respect to the suppression chamber, is always provided with a water seal from the containment.
7. The containment gas purge supply piping is Seismic Category I piping, which is pressurized to a pressure of approximately 120 psig by the Seismic Category I nitrogen supply system, thus precluding the possibility of leakage to the environment from the containment through these lines.
8. The DW inerting piping is isolated during normal plant operation and is used only during plant startup for DW purge/inerting.
9. The TIP nitrogen supply piping is Seismic Category I piping, which is pressurized to a pressure of approximately 120 psig by the Seismic Category I nitrogen supply system, thus precluding the possibility of leakage to the environment from the containment through these lines.

### 2.1.11 Design Inputs and Assumptions

Many of the design inputs and assumptions for the DBA radiological dose consequence analyses are different than those used in the current licensing basis. A listing of design inputs and assumptions used in AST analyses that are different from those used in the current licensing basis is provided in Appendix B.

## 2.2 Methodology

### 2.2.1 Accident Radiological Consequence Analysis

Analyses are prepared for the radionuclide release, transport, removal, and doses for the LOCA, FHA, CRDA, and MSLB.

The ORIGEN2 code is used to calculate plant-specific fission product inventories which bound the use of 24-month fuel cycles, operation at maximum licensed power, and current fuel designs. Bounding values of fission product activity are determined for each radionuclide in the DBA radiological analyses by considering enrichment and burnup. Fission product activities are calculated for immediately after shutdown and decayed for the required times. The core inventory is multiplied by a factor of 1.1 to provide margin for future fuel changes or power uprates.

The LocaDose code is used for the DBA dose calculations. LocaDose is a proprietary code for performing multi-node radiation transport and dose calculations. It consists of inter-related modules. The modules used in the AST analyses are the Activity Transport Program, Dose Calculation Program, and Gamma Source Generation Program. The transport program calculates time-dependent isotopic activities within nodes, based on production and removal terms specified for each node. The dose program calculates doses within nodes and at offsite locations. Doses are calculated using the dose conversion factors from FGRs 11 and 12. The source program converts the isotopic activities within nodes into integrated energy release values.

LocaDose has been utilized previously at HNP. It was used for analysis in support of the Extended Power Uprate approved in 1998, and recently in the license amendment to incorporate the use of KI in the current licensing basis as an interim measure (approved May 25, 2006). LocaDose has also been used in the approved AST license amendments for Surry and Catawba.

Shield-SG is used to refine the control room immersion dose calculated by LocaDose. Shield-SG is also used to calculate the dose in the MCR from the airborne activity within the TB. Shield-SG is a point-kernel computer program which allows three dimensional modeling of shielding problems.

STARNAUA is a Polestar Applied Technology, Inc. (Polestar) proprietary code that analyzes activity transport in particulate form. STARNAUA is used to model aerosol removal in containment, taking into account a number of processes including gravitational settling, diffusion of particles to surfaces within the containment volume, removal by sprays, and leakage of particles from the containment.

STARNAUA has been utilized in the AST license amendments for which the NRC issued Safety Evaluations for Perry and the Westinghouse AP600 and AP1000 designs. STARNAUA was utilized for Oyster Creek and Columbia, whose AST license amendment requests are under review.

Table 3 summarizes the various applications of Polestar's AST methodology for various sites and reactor designs.

### 2.2.2 Suppression Pool pH Control Analysis

The calculation methodology for suppression pool pH control is based on the approach outlined in NUREG-1465 and NUREG/CR-5950. Specifically, credit is taken for SPB addition to the suppression pool water as a result of SLC system operation. The pH of the suppression pool water is then calculated using the STARpH code. STARpH is a Polestar proprietary code that analyzes suppression pool pH as a function of time for systems containing radiolytically-produced strong acids and a variety of buffers and bases that may be used to control pH.

Calculations are performed to verify that sufficient SPB solution is available to maintain the suppression pool pH at or above 7 for 30 days post-accident. The design inputs are conservatively established to maximize the post-LOCA production of acids and to minimize the post-LOCA production and/or addition of bases. Other design input values such as initial suppression pool volume and pH are selected to minimize the calculated pH.

STARpH was used for applications receiving NRC Safety Evaluations for Perry, Vermont Yankee, Hope Creek, Waterford 3, and Browns Ferry. AST license amendment requests utilizing STARpH which are under review include Salem, Oyster Creek, and Columbia.

### 2.2.3 NUREG-0737 Analysis

An evaluation is performed to identify potential impacts of applying AST methodologies to NUREG-0737 items. This includes the following:

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

**Table 3. Polestar Methods for Activity Removal**

| | DW Spray | Main Condenser | STARNAUA | NRC Safety Evaluation |
|---|---|---|---|---|
| AP-600/1000 | | | X | X |
| Perry | X | | X | X |
| Browns Ferry | | X | | X |
| Vermont Yankee | X | X | | X |
| Columbia | X | | X | X |
| Oyster Creek | X | | X | |
| Hatch | X | X | X | |

Notes:
1. AP-600 and AP-1000 are advanced PWR designs (passive plants)
2. Perry design (Mark III containment) uses containment spray, not DW spray
3. The STARNAUA application at Columbia was used to adjust documented NRC Office of Nuclear Regulatory Research acceptance for Perry to account for DW spray
4. The NRC Safety Evaluation for Columbia was completed by the Dose Assessment Branch

## 2.3 Environmental Qualification

The radiation doses in the existing environmental qualification analyses were calculated using source terms determined by Technical Information Document (TID)-14844 methodology. Consistent with current regulatory allowance, the environmental qualification of equipment is bounded by, and will continue to be based on, TID-14844.

## 2.4 Atmospheric Dispersion Factors

### 2.4.1 Meteorological Data

The existing MCR $\chi/Q$ analysis was performed by use of one year meteorological data and ARCON95 in 1997 and has been approved by NRC. The postulated release locations from the reactor building (RB) and TB have been reviewed and approved by NRC. However, NRC recommended that any future analysis should be performed by use of ARCON96. Additionally, a Nuclear Energy Institute/NRC panel on "Control Room Habitability Analyses at Nuclear Power Plants" has recommended that 3-year meteorological data sets are more representative of the atmospheric dispersion conditions at a specific site than the use of one year data and should be used in future applications. Therefore, the $\chi/Q$ analysis for AST application is based on a set of 3-year meteorological data (1996-98) and is performed by use of ARCON96.

The maximum atmospheric dispersion factors (sec/m$^3$) for the MCR and TSC are shown in Table 4. The RB vent $\chi/Q$ values apply to any leakage prior to RB drawdown as well as stack bypass leakage after drawdown. The RB vent $\chi/Q$ values also apply to MSIV leakage in the TB.

Offsite atmospheric dispersion factors for containment, RB, and TB releases are shown in Table 5. The ground level $\chi/Q$ values apply to any leakage from the containment or RB prior to RB drawdown as well as stack bypass leakage after drawdown. The ground level $\chi/Q$ values also apply to MSIV leakage in the TB. The elevated $\chi/Q$ values apply to RB releases through the main stack after drawdown.

### 2.4.2 ARCON96

ARCON96 is a computer model developed by Pacific Northwest Laboratory for the NRC to estimate $\chi/Q$ values for onsite receptors near building structures. The model uses hourly meteorological data and recently developed methods to estimate atmospheric dispersion. ARCON96 is capable of evaluating ground-level, vent, and elevated releases. A vent release is a release that takes place through a roof-top vent with an uncapped vertical opening. This model also treats diffusion more realistically under low wind conditions than previous NRC-issued models.

**Table 4. Atmospheric Dispersion Factors - Main Control Room $\chi/Q$ (sec/m$^3$)**

| Release Point ⇒ | RB Vent | Main Stack |
|---|---|---|
| Receptor ⇒ | MCR Intake | MCR Intake |
| 0 – 2 hr | 1.41E-3 | 3.76E-6 |
| 2 – 8 hr | 1.08E-3 | 2.88E-6 |
| 8 – 24 hr | 4.70E-4 | 7.50E-7 |
| 1 – 4 day | 3.54E-4 | 7.67E-7 |
| 4 – 30 day | 2.67E-4 | 5.04E-7 |

Note: These $\chi/Q$ values are also applied to the TSC.

**Table 5. Atmospheric Dispersion Factors - Offsite $\chi/Q$ (sec/m$^3$)**

| Release Point ⇒ | Ground | | Elevated | |
|---|---|---|---|---|
| Receptor ⇒ | EAB | LPZ | EAB | LPZ |
| 0 – 2 hr | 3.1E-4 | 3.1E-4 | 1.7E-6 | 1.7E-6 |
| 2 – 8 hr | 3.1E-4 | 1.7E-4 | 1.7E-6 | 9.4E-7 |
| 8 – 24 hr | Not Applicable | 2.3E-5 | Not Applicable | 3.9E-7 |
| 24 – 96 hr | | 1.1E-5 | | 2.0E-7 |
| 96 – 720 hr | | 4.5E-6 | | 8.0E-8 |

Notes:
1. Although the EAB dose is calculated for a 2-hr period only, the $\chi/Q$ values are applied for 8 hr to determine the worst-case 2-hr dose.
2. The elevated $\chi/Q$ values apply to RB releases through the main stack after drawdown. The ground level $\chi/Q$ values apply to other releases from the containment, the RB, and the TB.

## 2.5 Accident Radiological Consequence Analyses

The DBA accident analyses documented in Chapter 15 of the HNP FSAR that could potentially result in control room and offsite doses are addressed using methods and input assumptions consistent with the AST. The following DBAs are addressed:

- LOCA, FSAR Section 15.3.3
- FHA, FSAR Section 15.3.5
- CRDA, FSAR Section 15.3.2
- MSLB Accident, FSAR Section 15.3.4

Radiological consequences in the analyses have changed due to the impact of the characteristics of the AST itself and licensing basis changes that are being made concurrent with the AST implementation. Analyses are performed per RG 1.183. Documentation of conformance to RG 1.183 is included as Appendix A to this enclosure. The results of the radiological dose consequence analyses are evaluated to confirm compliance with the acceptance criteria presented in 10 CFR 50.67 and GDC 19 of 10 CFR 50, Appendix A.

Radiological dose consequences are evaluated for individuals located at two offsite locations, the exclusion area boundary (EAB) and the outer boundary of the low population zone (LPZ). Dose consequences are also evaluated for personnel occupying the MCR and the TSC. Although RG 1.183 does not address TSC dose limits, the dose limits that apply to the control room are assumed to apply to the TSC as well.

### 2.5.1 Common Inputs and Assumptions

The design inputs, parameters, and assumptions that are common to multiple DBAs are presented in Table 6.

**Table 6. Common DBA Radiological Consequence Analyses Inputs and Assumptions**

| Input / Assumption | Value |
|---|---|
| Reactor Power | 2818 MWt (current licensed rated thermal power level times an uncertainty factor of 1.005) |
| Core Inventory | The equilibrium core inventory of fission products per unit power (Ci/MWt) has been generated using the ORIGEN2 computer program based on a 24-month fuel cycle. The inventory is limited to the radionuclide groups and elements specified in RG 1.183, Table 5. In addition, a 10% margin is incorporated into the core inventory to allow for future fuel changes or power uprates. The core inventory (without the 10% margin) is given in Table 7. |
| **Volumes** | |
| MCR Free Volume | 9.35E4 $ft^3$ |
| TSC Free Volume | 1.56E4 $ft^3$ |
| TB Free Volume | 6.50E6 $ft^3$ |
| Turbine / Condenser Volume | 1.72E5 $ft^3$ (combined volume of low-pressure turbine and condenser) |
| **MCR / TSC Ventilation** | |
| MCR Filtered Intake Rate | The design value is 400 cfm, however, to allow for the potential need for less flow to pressurize the MCR, the filtered intake rate conservatively assumed in the analyses is 250 cfm. |
| MCR Recirculation | 2,100 cfm |
| MCR Filter Efficiency (Intake and Recirculation) | 95% for all radionuclides except noble gases |
| MCR Unfiltered Inleakage | 115 cfm (for LOCA, CRDA, and MSLB) |
| MCR Ingress / Egress | 2 one-way trips per day, lasting 2 min (each way) |
| TSC Filtered Intake Rate | 500 cfm |
| TSC Recirculation | 500 cfm |
| TSC Filter Efficiency (Intake and Recirculation) | 90% for all radionuclides except noble gases |

| Input / Assumption | Value |
|---|---|
| TSC Unfiltered Inleakage | 10,000 cfm |
| **Breathing Rates and Occupancy Factors** | |
| Breathing Rate (MCR, TSC, TB) | 3.5E-04 $m^3$/sec |
| Breathing Rate (Offsite) | 0-8 hr: 3.5E-04 $m^3$/sec<br>8-24 hr: 1.8E-04 $m^3$/sec<br>1-30 days: 2.3E-04 $m^3$/sec |
| MCR and TSC Occupancy Factors | 0-1 day: 100%<br>1-4 days: 60%<br>4-30 days: 40% |
| **TB Ventilation** | |
| TB Fans – Time to Start | 9 hr after initiation of the accident (credited operator action). TB exhaust is credited for the LOCA, CRDA, and MSLB. |
| TB Fan Exhaust Rate | 15,000 cfm (capacity of one fan) unfiltered release to the environment via the RB vent. TB exhaust is credited for the LOCA, CRDA, and MSLB. |

**Table 7. Core Inventory**

| | Isotope | Source (Ci/MWt) | | Isotope | Source (Ci/MWt) | | Isotope | Source (Ci/MWt) | | Isotope | Source (Ci/MWt) |
|---|---|---|---|---|---|---|---|---|---|---|---|
| Iodines | I-129 | 1.23E-03 | Tellurium Metals | Se-79 | 1.66E-02 | Noble Metals | Ru-103 | 4.25E+04 | Lanthanides | Sm-151 | 1.90E+01 |
| | I-130 | 1.05E+03 | | Sb-124 | 4.54E+01 | | Ru-105 | 2.97E+04 | | Sm-153 | 1.37E+04 |
| | I-131 | 2.72E+04 | | Sb-125 | 6.10E+02 | | Ru-106 | 1.70E+04 | | Eu-152 | 4.70E-01 |
| | I-132 | 3.93E+04 | | Sb-126 | 3.58E+01 | | Rh-103m | 3.83E+04 | | Eu-154 | 3.93E+02 |
| | I-133 | 5.52E+04 | | Sb-126m | 1.36E+01 | | Rh-105 | 2.72E+04 | | Eu-155 | 2.75E+02 |
| | I-134 | 6.05E+04 | | Sb-127 | 2.98E+03 | | Rh-106 | 1.82E+04 | | Eu-156 | 5.09E+03 |
| | I-135 | 5.16E+04 | | Sb-129 | 8.85E+03 | | Pd-107 | 4.11E-03 | | Am-241 | 6.86E+00 |
| | I-136 | 2.45E+04 | | Te-125m | 1.33E+02 | | Pd-109 | 9.40E+03 | | Am-242 | 2.85E+03 |
| | I-137 | 2.39E+04 | | Te-127 | 3.00E+03 | Lanthanides | Y-89m | 6.27E-04 | | Am-242m | 9.30E-01 |
| | I-138 | 1.18E+04 | | Te-127m | 4.00E+02 | | Y-90 | 3.20E+03 | | Am-243 | 7.46E-01 |
| Noble Gases | Kr-83m | 3.30E+03 | | Te-129 | 8.71E+03 | | Y-90m | 4.61E-01 | | Cm-242 | 1.68E+03 |
| | Kr-85 | 3.78E+02 | | Te-129m | 1.29E+03 | | Y-91 | 3.23E+04 | | Cm-243 | 8.99E-01 |
| | Kr-85m | 6.92E+03 | | Te-131 | 2.41E+04 | | Y-91m | 1.83E+04 | | Cm-244 | 1.05E+02 |
| | Kr-87 | 1.32E+04 | | Te-131m | 3.94E+03 | | Y-92 | 3.44E+04 | | Cm-245 | 1.00E-02 |
| | Kr-88 | 1.86E+04 | | Te-132 | 3.85E+04 | | Y-93 | 4.00E+04 | | Cm-246 | 1.65E-03 |
| | Kr-89 | 2.26E+04 | | Te-133 | 3.24E+04 | | Y-94 | 4.05E+04 | Halogens | Br-82 | 1.81E+02 |
| | Xe-131m | 3.03E+02 | | Te-133m | 1.99E+04 | | Y-95 | 4.36E+04 | | Br-83 | 3.30E+03 |
| | Xe-133 | 5.27E+04 | | Te-134 | 4.53E+04 | | Zr-93 | 9.53E-02 | | Br-84 | 5.69E+03 |
| | Xe-133m | 1.58E+03 | Strontium, Barium | Sr-89 | 2.49E+04 | | Zr-95 | 4.77E+04 | | Br-85 | 6.83E+03 |
| | Xe-135 | 1.89E+04 | | Sr-90 | 3.01E+03 | | Zr-97 | 5.00E+04 | | Br-87 | 1.11E+04 |
| | Xe-135m | 1.09E+04 | | Sr-91 | 3.15E+04 | | Nb-93m | 8.21E-03 | | Br-88 | 1.19E+04 |
| | Xe-137 | 4.81E+04 | | Sr-92 | 3.42E+04 | | Nb-95 | 4.79E+04 | Cerium Group | Ce-141 | 4.48E+04 |
| | Xe-138 | 4.52E+04 | | Sr-93 | 3.90E+04 | | Nb-95m | 3.37E+02 | | Ce-143 | 4.13E+04 |
| Alkali Metals | Rb-86 | 7.06E+01 | | Sr-94 | 3.69E+04 | | Nb-97 | 5.04E+04 | | Ce-144 | 3.69E+04 |
| | Rb-88 | 1.89E+04 | | Sr-95 | 3.43E+04 | | La-140 | 4.88E+04 | | Np-237 | 1.61E-02 |
| | Rb-89 | 2.42E+04 | | Ba-136m | 3.60E+02 | | La-141 | 4.47E+04 | | Np-238 | 1.48E+04 |
| | Rb-90 | 2.34E+04 | | Ba-137m | 3.92E+03 | | La-142 | 4.31E+04 | | Np-239 | 5.71E+05 |
| | Cs-134 | 6.83E+03 | | Ba-139 | 4.91E+04 | | La-143 | 4.11E+04 | | Pu-236 | 2.74E-02 |
| | Cs-134m | 1.65E+03 | | Ba-140 | 4.73E+04 | | Pr-143 | 4.04E+04 | | Pu-237 | 1.04E-01 |
| | Cs-135 | 2.35E-02 | | Ba-141 | 4.45E+04 | | Pr-144 | 3.71E+04 | | Pu-238 | 1.31E+02 |
| | Cs-136 | 2.18E+03 | Noble Metals | Co-58 | 1.48E+02 | | Pr-144m | 4.44E+02 | | Pu-239 | 1.29E+01 |
| | Cs-137 | 4.14E+03 | | Co-60 | 4.27E+02 | | Nd-147 | 1.80E+04 | | Pu-240 | 1.78E+01 |
| | Cs-138 | 5.02E+04 | | Mo-99 | 5.15E+04 | | Pm-147 | 4.53E+03 | | Pu-241 | 5.23E+03 |
| | Cs-139 | 4.75E+04 | | Tc-99 | 5.24E-01 | | Pm-148 | 7.82E+03 | | Pu-242 | 6.19E-02 |
| | | | | Tc-99m | 4.53E+04 | | Pm-148m | 1.16E+03 | | Pu-243 | 1.06E+04 |
| | | | | Tc-101 | 4.63E+04 | | Pm-149 | 1.60E+04 | | | |
| | | | | | | | Pm-151 | 5.49E+03 | | | |

Note: In calculating doses, the progeny of these isotopes is also included and the activities are increased by 10% to allow for future fuel changes or power uprates.

### 2.5.2 Loss-of-Coolant Accident (LOCA)

The radiological dose consequences to occupants of the MCR and the TSC and to persons located at the EAB and LPZ following a postulated LOCA are determined using AST assumptions and methodology in accordance with RG 1.183.

2.5.2.1 Inputs and Assumptions

*Release from Core*

In accordance with RG 1.183, Table 4, the core activity is assumed to be released into the containment in two phases: gap activity release (starts at 2 min and lasts 30 min) and early in-vessel release (starts at the conclusion of the gap activity release phase and lasts for 90 min). The core fractions of the radionuclide groups that are released during each phase are based on RG 1.183, Table 1. The release rates are shown in Table 8.

*Chemical Form*

By adding boron to the suppression pool as a credited operator action, the pH of the torus water is maintained at a value of 7 or higher. This is based on full mixing of the boron within the pool water within the first 24 hr of the accident. With this pH, the chemical form of iodine released to the containment is assumed to be 95% cesium iodide (CsI), 4.85% elemental, and 0.15% organic per RG 1.183, Appendix A, Section 2. With the exception of elemental and organic iodine and noble gases, all isotopes are assumed to be in particulate form.

*Containment Volume and Mixing*

It is assumed that the activity released from the fuel is instantaneously and homogeneously mixed throughout the free air volume of the DW (RG 1.183, Appendix A, Section 3.1). It is also assumed that the DW activity starts flowing into the torus air volume at the end of the early in-vessel release phase as shown in Table 9.

*Containment Activity Removal*

Credit is taken for the reduction of airborne activity due to natural deposition and sprays (RG 1.183, Appendix A, Sections 3.2 and 3.3). The removal rates for elemental iodine and particulates within the DW due to natural deposition (sedimentation) and sprays (with the sprays assumed to start at 15 min as a credited operator action) are shown in Table 10. The removal rates are based on credit for one residual heat removal (RHR) pump. The manual initiation of DW sprays is a required operator action that is initiated based upon reaching a DW dose rate of 200,000 rem/hr. Analysis of the DW immersion dose rate following a DBA LOCA shows that this dose rate is reached within 15 min of accident initiation.

Modeling with STARNAUA

Aerosol removal in containment is governed by a number of processes modeled by the STARNAUA computer code including gravitational settling (sedimentation) and removal by sprays. In addition, agglomeration (coagulation) of particles is modeled; removal by sprays may be considered a special case of agglomeration.

Agglomeration is more pronounced when the number density of particles in the containment atmosphere is large. It is apparent from Stokes Law that larger particles are removed more efficiently than smaller ones (both for sedimentation and for spray removal); therefore, agglomeration can substantially increase the removal rate. Because large particles are more readily removed than smaller ones, the particle size distribution gets further depleted of large particles with time. Agglomeration mitigates this trend, tending to be a source of large particles.

As noted, an important special case of interest occurs when containment sprays are present. Here the agglomeration takes place between the very large spray droplets and the aerosol particles, which results in a very efficient process for removing the particles.

Sedimentation is always an important removal mechanism for aerosol particles and is often the predominant one, even in the absence of sprays. When sprays are present, sedimentation of the compound spray droplet/particle accounts for almost all of the aerosol removal.

Sedimentation of aerosols (as opposed to that of spray droplets) is well understood in terms of the Stokes-Cunningham law which gives the terminal settling velocity for a single particle of actual radius $r_p$ as:

$$v_s = \frac{2Cg(\rho_p - \rho_{gas})r_p^2}{9\mu_{gas}}$$

where:

$v_s$ = settling velocity, cm/sec
$g$ = gravitational acceleration, 980 cm/sec$^2$
$\rho_p$ = density of the particle, g/cm$^3$
$\rho_{gas}$ = density of air (or the containment atmosphere), g/cm$^3$
$\mu_{gas}$ = viscosity of air (or the containment atmosphere), g/cm-sec
$C$ = Cunningham slip factor = $1 + \frac{\lambda}{r_p}\left(1.246 + 0.42\exp(-\frac{0.87r_p}{\lambda}\right)$
$\lambda$ = gas mean free path, cm

The Stokes law expression above is valid for particles of radius less than ~ 50 microns (i.e., aerosols) which adequately covers the particle size range of interest. (It does not apply to spray droplets, which are considerably larger and are treated differently.)

Settling and spray removal rates are strongly dependent on the size (and material density) of the aerosol particles. In STARNAUA, the aerosol population is characterized by a size distribution which evolves in time from its initial (source) function to a time-dependent distribution as particles of different sizes are added, agglomerate, and/or are removed at different rates. The initial source size distribution is assumed to be characteristic of the aerosol released into the containment.

In so-called "discrete" codes (including STARNAUA) the size distribution is defined by choosing appropriate minimum and maximum particle sizes and dividing the interval between the two into a number of "bins." The initial size distribution is then entered as the fraction of the total number of particles assigned to each bin. The population of each bin is then followed as a function of time. The larger the number of bins, the more accurately the distribution will be represented, but at the cost of increased computing time. The HNP analysis for DW spray was conducted using 30 bins covering 4.54E-03 microns to 90.8 microns aerodynamic radius for the initial distribution.

It has been observed that many aerosol distributions (number of particles, N, as a function of particle radius, r), both those occurring naturally and those resulting from industrial or other processes of human origin, are of the log-normal type. It should be noted that such a distribution is completely defined by two parameters: $(\ln(r))_m$ and $\sigma_g$. $(\ln(r))_m$ is the mean value of ln(r) for the aerosol population. To characterize the initial (source) distribution, STARNAUA replaces $(\ln(r))_m$ by the closely related parameter $r_g$, where $r_g$ is the (geometric) mean particle radius ($\ln(r_g) = (\ln(r))_m$).

Polestar analyzed the results of a number of large-scale fuel melt experiments in order to obtain values of $r_g$ and $\sigma_g$. The results were then averaged to obtain what Polestar believes to be the most representative values of the source size distribution in a reactor fuel melt accident. The representative values are $r_g = 0.22$ μm and $\sigma = 1.81$ ($\sigma_g = 0.5933$). These are the values that are normally input into STARNAUA calculations at Polestar (except for a small correction due to the void fraction of the particles).

An important consideration in sedimentation arises from the fact that the aerosol particles are not considered to be solid, but are assumed to have void fractions that are filled with gas if the containment atmosphere is dry or filled with water if the atmosphere is at or near saturation. For plants with sprays operating, it is generally assumed that the voids are water-filled.

The mechanisms that contribute to the spray collection efficiency modeled in STARNAUA include interception, impaction, Brownian diffusion of aerosol particles to the droplet, and diffusiophoretic deposition of particles to the droplets if the thermal-hydraulic conditions result in steam condensation on the droplets. The latter effect is neglected for HNP. The overall spray collection efficiency is the sum of the individual efficiencies of these processes, which are dependent on both the droplet and the particle sizes.

Thermal-Hydraulic Considerations

The density and viscosity of the containment atmosphere are functions of input thermal-hydraulic conditions in the containment (temperature and gas composition (steam/nitrogen ratio)), which will vary with time. STARNAUA contains function statements that yield these quantities at each time step in the calculation. Three important phases in the assumed HNP accident sequence are:

1. Prior to DW spray actuation (0 to 15 min, with the activity release beginning at 2 min)
2. Release phase with sprays operating (15 min to 122 min)
3. Post-release phase (122 min to 24 hr)

During Phase (1), the only removal phenomenon that is credited is sedimentation. For sedimentation, the higher the gas temperature, the less effective the aerosol sedimentation. Consequently, for purpose of conservatism, the thermal-hydraulic inputs for this phase are chosen at a point in time when the DW temperature is the highest.[2]

During Phases (2) and (3), DW sprays are operating. Unlike aerosol sedimentation (where the

[2] Due to the conservative modeling for containment activity removal, the values for temperature and pressure used by the STARNAUA code (and presented in Table 11 and Table 12) are different from the temperature and pressure values presented in Table 16.

conservatism of high temperature may correspond to a low gas density), the higher the gas density, the less effective the spray removal. Therefore, the thermal-hydraulic inputs for these two phases are chosen at a point in time when the DW pressure and gas density is the highest. The DW thermal-hydraulic data used by STARNAUA are shown in Table 11, Table 12, and Table 13.

Mass of Inert Aerosol Release

In addition to fission products released as aerosols, non-fission product fuel and structural ("inert") material aerosols are also released. Although these do not contribute significantly to radiation exposure, their presence in the total aerosol is important, since they do contribute to the aerosol number density in the containment atmosphere and take part in aerosol agglomeration. Thus, they influence the removal rates of the fission product aerosols from the containment atmosphere. It is thus essential to include them in the aerosol source term in STARNAUA calculations. The ratio of structural material to fission product aerosols should be at least 2.4. This value is considered to be conservative, i.e. it will result in less aerosol removal than a larger value would. For HNP, an even more conservative value of 1:1 was used.

In a STARNAUA calculation the fission product aerosol release in each release period for each fission product is determined on the basis of its core inventory at the time of the accident and its release fraction from the core. The fission product release rates are summed and for the in-vessel release period the inert release rate is taken as equal to the sum of the total fission product releases (gap plus in-vessel release periods) times the assumed inert/fission product ratio. It is assumed that no inert release occurs during the gap release period.

The removal of elemental iodine is assumed to occur at the same rate and with the same degree of completeness as particulate. This is based on the propensity for elemental iodine to adsorb onto surfaces (in this case, the large surface area of the dispersed particulate). Once the iodine is dissolved in the spray water, for a suppression pool pH of 8.3 at 24 hr, the point in time when spray credit ceases (see Section 2.6), the ratio of iodine concentration in the liquid phase to that in the gas will exceed 30,000. For a primary containment gas-to-liquid volume of approximately 3 for HNP, this means that not more than 0.01% of the total iodine would remain airborne as elemental iodine at that time, and most likely, even less. This is negligible considering the residual 0.15% iodine airborne in organic form. With the suppression pool pH being greater than 7.7 even at 30 days (well above neutral pH), re-evolution of elemental iodine later in the accident does not need to be considered.

*Containment Leakage*

The primary containment (DW and torus air) is assumed to leak at the peak pressure TS leak rate of 1.2% weight per day for the first 24 hr of the accident, reducing by 40% from 24 to 72 hr, and by 50% thereafter, as justified in Section 2.5.2.2. Assuming the activity within the containment to be uniformly distributed, the volumetric leak rate is the same as the mass leak rate. Starting with the gap activity release phase, all the leakage from the primary containment (excluding MSIV leakage) enters the secondary containment except for 2% that is assumed to bypass the secondary containment. The bypass leakage is assumed to be directly to the environment at ground level in evaluating offsite and TSC doses and into the condenser in evaluating MCR doses.

*Secondary Containment*

It is assumed that the RB (secondary containment) draws down to negative pressure within 2 min of the start of the accident. After secondary containment drawdown, RB activity is released to the environment through the plant stack at the maximum TS rate of 4,000 cfm per unit. It is possible for two standby gas treatment system (SGTS) fans to be in operation at the same time, taking suction from one RB and the refueling floor. In order to maximize the release to the environment, it is conservatively assumed that the entire flow of 8,000 cfm from two fans is from a single RB. The release is processed by the SGTS filters with an efficiency of 95% for all isotopes except noble gases. The activity within the RB is assumed to be uniformly distributed within 50% of the volume (RG 1.183, Appendix A, Section 4.4). The total volume of the RB is obtained by adding the individual compartment volumes, conservatively neglecting the common refueling floor to minimize the dilution.

*ESF Leakage*

In modeling engineered safety feature (ESF) leakage, it is conservatively assumed that all the isotopes that are released to the containment except noble gases are instantaneously transported to the torus water at the onset of the gap activity release phase and mixed uniformly (RG 1.183, Appendix A, Section 5.1). Although there is no TS limit on ESF leakage, a conservatively high leakage rate of 10 gpm (1.34 cfm) is assumed to start at the initiation of sprays, lasting for the duration of the accident. With the torus water temperature below 212°F, it is assumed that 10% of the iodine in the ESF leakage becomes airborne inside the RB while all other elements remain in the water; of the iodine that becomes airborne, 97% is assumed to be elemental and 3% organic (RG 1.183, Appendix A, Section 5.3 to 5.6).

*MSIV Leakage*

It is assumed that the maximum leakage from all four MSLs is 100 scfh,[3] with no limit on the leakage per line. It is postulated that the inboard MSIV on one of the four steam lines fails to close, thus creating an unrestricted flow path to the outboard MSIV. It is assumed that the full leakage of 100 scfh is through this failed line with no MSIV leakage through any of the three intact lines. This is conservative as there is less activity removal due to deposition within a line with a failed MSIV than with intact MSIVs. The leakage rate is reduced by 40% at 24 hr and by 50% at 72 hr, as justified in Section 2.5.2.2.

The source of the leakage is the airborne activity in the DW (RG 1.183, Appendix A, Section 6.1). The MSIV leakage could take place in either the RB or the TB. Since a leakage in the RB would have the benefits of filtration and dispersion due to an elevated release through the stack after RB drawdown, it is assumed that all MSIV leakage occurs in the TB. Since the MCR is located within the TB, it is conservative to calculate MCR doses assuming holdup in the TB as this provides a direct inleakage pathway from the TB to the MCR.

*Main Steam Line Activity Removal*

The calculation of aerosol removal in the MSLs is also accomplished with STARNAUA.[4] It is assumed that particulate in the portion of the MSL (1) between the inboard and the outboard MSIVs for three lines and (2) between the outboard MSIV and the main stop valve for all lines is subject to removal by deposition, as allowed by RG 1.183, Appendix A, Section 6.5. Since it may be assumed that all MSIV leakage occurs in a single line (the limiting case being the line with only one MSIV closed), the limiting case involves deposition only between the outboard MSIV and the main stop

[3] 100 scfh represents the maximum allowable MSIV leakage at reduced test pressure. For testing at or above accident pressure, the maximum allowable MSIV leakage is 144 scfh. See Section 2.5.2.2 for further discussion.
[4] The methodology of AEB 98-03 (Reference 2) is not utilized.

valve. Table 14 shows the time-dependent deposition rates for that limiting case. The particle size distribution used as input for the MSL analysis is that representing airborne activity from the DW analysis.

Because sedimentation is minimized by the assumption of high temperature, the steam line is conservatively assumed to remain at its maximum temperature (551°F) during the full duration of the analysis. Only horizontal runs of steam line are credited for sedimentation, and only the projected area of the steam line is used as sedimentation area.

It is also assumed that particulate mass and activity and elemental iodine activity are reduced by a factor of two due to particle impaction at the inboard or outboard MSIV, whichever is the first closed valve encountered (as noted, three lines are assumed to have both valves closed). Once the particulate enters the steam line beyond the first closed MSIV, however, no further elemental iodine removal is considered in the steam lines or main condenser. This is very conservative, because even if re-evolution from the particulate surfaces were to occur in the hot, dry conditions of the steam line, some deposition and retention would be expected on the metal surfaces in those volumes, as well.

*Condenser and Turbines*

Most of the MSIV leakage reaches the condenser and the low pressure turbine while a small fraction bypasses the condenser/low pressure turbine and is released through the high pressure turbine (HPT). The bypass fraction is calculated to be 0.005 per the methodology given in Reference 3. It is assumed that particulates in the condenser are removed by sedimentation as shown in Table 15. Although the entering flow is 100 scfh less the bypass fraction, the condenser leak rate is assumed to be 100 scfh during the first 24 hr, reducing by 40% at 24 hr and by 50% at 72 hr.

Particulate removal by sedimentation in the condenser is credited for both MSIV leakage and secondary containment bypass leakage.

*Turbine Building*

In calculating MCR doses, it is assumed that releases from the condenser and the turbines are into the TB, where they are available for direct inleakage into the MCR. It is assumed that the released activity is uniformly mixed within the volume of TB elevation 164 ft floor, which is open to both units.

*Atmospheric Dispersion*

All releases from the TB are assumed to be at ground level through the RB vent. MCR atmospheric dispersion factors ($\chi/Q$) are shown in Table 4. The MCR values are applied to the TSC since the MCR values are bounding. The EAB and LPZ $\chi/Q$ values are shown in Table 5. The 0-2 hr EAB $\chi/Q$ values are applied for the first 8 hr to ensure that the EAB dose is calculated over a 2-hr period that yields the maximum dose per RG 1.183, Section 4.1.5.

*Accident Duration*

MCR, TSC, and LPZ doses are calculated assuming a release duration and exposure time of 30 days per RG 1.183, Table 6. The EAB dose is calculated over a two-hr period which yields the maximum dose per RG 1.183, Section 4.1.5.

*Credited Operator Actions*

The following actions taken by operators are credited in the analysis:

1. MSIV ALT Pathway – Lining up the MSIV ALT pathway by opening a valve to establish the pathway and closing boundary valves to direct MSIV leakage to the condenser for holdup and retention of activity in MSL piping and the condenser.
2. Addition of pH Buffering Agent – Addition of SPB to the suppression pool via operation of the SLC system to maintain suppression pool pH at 7 or higher for the duration of the accident, thereby precluding the re-evolution of iodine from the suppression pool.
3. DW Sprays – Initiation of DW sprays based on radiation levels in the DW to help remove airborne particulate in the DW and reduce DW temperature and pressure.
4. TB Fans – Initiation of one TB exhaust fan within 9 hr of the start of the accident to remove activity from the TB.

2.5.2.2 Method of Evaluation

Doses for the LOCA are calculated using the guidance provided in the main body and Appendix A of RG 1.183. The computer codes LocaDose and Shield-SG are utilized to calculate doses. Descriptions of these codes are provided in Section 2.2.1. The inputs and assumptions used to determine offsite, MCR, and TSC doses are described in the previous section. Assumptions requiring further elaboration are discussed below.

Activity transport models are developed for both the RB and the TB. There are three main activity release pathways:

- Containment leakage – This includes leakage of airborne activity in the DW and the torus air space. Initially any leakage is assumed to be directly to the environment at ground level. After RB drawdown to negative pressure, the leakage is assumed to be processed by the SGTS and released through the plant stack except for a small fraction that bypasses the SGTS.
- MSIV leakage – This leakage reaches the condenser via the steam lines. The condenser is then assumed to leak to the TB and eventually to the MCR and the environment.
- ESF leakage – This occurs in the RB after the DW sprays have been initiated and water from the torus is recirculated back into the DW.

To determine offsite doses, two models are developed. The first calculates doses from elevated releases from the RB via the SGTS. The second model is for ground releases, which include RB activity bypassing the SGTS and MSIV leakage activity leaking from the condenser. The ground level releases are through the RB vent. Total offsite doses are a combination of doses from elevated releases and ground releases.

In evaluating TSC doses, RB activity is released to the environment through the stack via the SGTS and the RB vent (for activity that bypasses the SGTS). MSIV leakage activity leaking from the condenser is released to the environment via the RB vent. Activity released to the environment reaches the TSC via outside air intake and unfiltered inleakage.

For MCR doses, RB activity is released to the environment through the stack via the SGTS. RB bypass leakage is assumed to go into the condenser, where it leaks into the TB. MSIV leakage activity also collects in the TB via leakage from the condenser. Beginning at 9 hr after the accident, TB air is exhausted at a rate of 15,000 cfm. Activity released to the environment reaches the MCR via an outside air intake at a rate of 250 cfm. TB activity also leaks directly into the MCR (unfiltered inleakage).

*SLC Injection*

HNP takes credit for the SLC system for the injection of a sufficient quantity of SPB solution into the reactor vessel, and ultimately mixing in the suppression pool, to meet the requirement for maintaining pH at or above 7. The mixing scenario is as follows:

1. It is conservatively assumed that there is no SLC injection during the first 2 hr of the accident. Applicable plant procedures will be revised as needed to ensure that SLC injection is initiated within 2 hr of accident initiation.
2. Starting at 2 hr, SLC injection is initiated and lasts for up to 1.5 hr.
3. After SLC injection is completed at 3.5 hr, the core spray (CS) system floods the reactor vessel. Assuming a single CS pump at a minimum flow rate of 4,000 gpm is used to fully replenish the entire primary system volume, the time required for the fill is 0.5 hr.
4. After the reactor vessel is filled at about 4 hr, the excess liquid will spill out of the break and reach the suppression pool, thereby initiating the mixing of the borate solution within the pool.
5. Suppression pool water is recirculated through the reactor vessel and the DW via the CS and RHR systems, respectively, resulting in further mixing. Based on the suppression pool volume, the flow rate of a single RHR pump, and the CS flow rate, the time required for one turnover of the suppression pool water is approximately 1 hr.

Therefore, it takes about 5 hr to complete the first turnover of the suppression pool water. As each subsequent turnover of the suppression pool water requires about 1 hr, there will be about 20 turnovers during the first 24 hr. It is expected that a few turnovers of the pool volume is all that is required to fully mix the SPB solution.

The calculation of the suppression pool pH is described in Section 2.6.

*Reduction in Containment and MSIV Leakage Rates*

Leakage may be reduced after the first 24 hr, if supported by plant configuration and analyses, to a value not less than 50% of the TS leakage rate (RG 1.183, Appendix A, Section 3.7 and 6.2).

For pressurized water reactors (PWRs), RG 1.183 allows the containment leakage rate to be reduced by 50% without further justification. The containment leakage rate is dependent on the containment pressure. For both BWRs and PWRs, after the initial transient conditions following a LOCA, the containment pressure steadily decreases with time unless some action is taken to increase the pressure.

Some BWRs, including HNP, have containment atmosphere dilution (CAD) systems to control the hydrogen and oxygen levels inside the containment. The CAD system works by injecting nitrogen into the containment, which has the potential to increase pressure. The severe accident guidelines and emergency operating procedures for HNP call for the DW and the torus to be vented while nitrogen is being injected such that the pressure does not increase. Therefore, without a pressure increase associated with the CAD system, the HNP containment leakage rate would be expected to decrease in the same manner as a BWR without a CAD system or a PWR.

The containment and MSIV leakage rates are both based on the peak DW pressure and temperature. The volumetric flow rates are calculated at these peak conditions as well as the later time steps, when the pressure and temperature have decreased. By comparing the flow rates at different time steps, the reduction in leakage rate can be calculated at 24 hr or any other time step. Since the flow rate through the MSLs is driven by the pressure difference between the DW upstream and the condenser downstream, it is reasonable to assume that the MSIV leakage rate will be reduced by the same magnitude as containment leakage.

It is assumed that the containment and MSIV leakage paths are sufficiently restrictive that the flows are unchoked and may be treated as incompressible. The containment leakage is into the RB while the MSIV leakage is into the TB via the condenser. For both leakages, the downstream pressure in the RB and TB is assumed to be atmospheric at 14.7 psia.

The use of sprays reduces the DW pressure from a peak value of 65.5 psia in Unit 1, and a peak value of 62.0 psia in Unit 2. A summary of parameters for containment leakage reduction is shown in Table 16.

The MSIV leakage rate is based on a peak pressure of 61.6 psia and a peak temperature of 340°F. The saturation steam pressure at 340°F exceeds 61.6 psia, meaning the steam is superheated. In applying the flow equation at 0 hr, increasing the density causes the flow rate to decrease, which results in higher flow ratios at 24 and 72 hr. Hence, to minimize the initial flow rate, the maximum possible air density is added to the superheated steam density. The air pressure used to calculate the air density is taken as the difference between 61.6 psia and the design DW pressure of 62 psig or 76.7 psia. A summary of parameters for MSIV leakage rate reduction for both units is shown in Table 17.

As shown in Table 16 and Table 17, the containment and MSIV leakage rates may be reduced to 60% of the initial values at 24 hr and to 50% at 72 hr.

*MSIV Leakage*

The mass flow rate of 100 scfh (maximum allowable MSIV leakage at reduced test pressure) is converted to a true volumetric flow rate (cfh or cfm) for the appropriate conditions:

| | |
|---|---|
| DW to MSL | 100 scfh = 49.7 cfh = 0.828 cfm |
| MSL to Condenser/HPT | 100 scfh = 263 cfh = 4.38 cfm |

The methodology used in performing this conversion is as follows:

1. Determine an orifice size corresponding to MSIV leakage under test conditions
2. Determine a volumetric leak rate per unit of orifice surface area under accident conditions
3. Multiply the results of 1 and 2 above to determine volumetric leak rate under accident conditions

For MSIV testing conducted at peak containment pressure or greater, the MSIV allowable leakage limit is 144 scfh for both units. The conversion of the 144 scfh to a true volumetric flow rate (cfh or cfm) for the appropriate conditions results in the same (or lower) cfh and cfm results as those given above for the reduced pressure testing case. That is, the leak path size corresponding to 100 scfh at reduced test pressure is equal to or larger than the leak path size corresponding to 144 scfh at peak containment pressure.

The flow out of the MSL is apportioned between the condenser and the HPT based on the bypass fraction of 0.005. Although the entering flow is slightly less than 100 scfh, the flow out of the condenser is assumed to be 100 scfh. The flow rate out of the HPT is assumed to be equal to the entering flow. The leak rates are reduced by 40% at 24 hr and 50% at 72 hr. The flows are summarized in Table 18.

*MCR Dose due to Airborne Activity in Turbine Building*

In calculating doses within regions, LocaDose only considers activities within the region. Although the MCR has 2 ft thick concrete walls, the radiation shine dose from the airborne activity within the TB could be significant because the MCR is located in the TB. The Shield-SG computer program is used to calculate the MCR dose due to TB activity.

A dose of 2.38E-03 rem is calculated for a conservatively low value of TB exhaust rate, and therefore the highest TB activity. This external shine dose from TB activity is conservatively applied to the MCR dose evaluation.

*MCR Dose Due to Other External Sources*

In addition to the dose contributions from the MCR air, ingress and egress through the TB, and TB air, the shine from other external sources is evaluated. Other external shine sources considered are secondary containment, the cloud outside the TB, MSLs, condenser, and MCR filters. Analysis has determined that the shine dose from these sources is estimated to be 0.03 rem TEDE.

*MCR Ingress/Egress Dose*

Since the MCR is located in the TB, the MCR operator would walk through the TB when entering or leaving the MCR. A conservative maximum walking distance through the TB is estimated from TB dimensions. Using this distance, the transit time through the TB is estimated based on a walking speed of 3 miles/hr. An additional time of 45 sec for using stairwells and opening doors is added, and the total transit time is assumed to be 2 min.

The ingress/egress dose is calculated by determining the average TB dose rate during a time interval and multiplying by the exposure duration. Assuming two one-way trips per day, the doses from all trips over the 30-day duration of the accident are added to obtain the total ingress/egress dose.

#### 2.5.2.3 Results

Post-accident doses are the result of the following activity considerations:

1. Primary to secondary containment leakage – Activity is released directly into secondary containment and filtered by the SGTS prior to elevated release through the plant stack, except for a small amount that bypasses the SGTS. Prior to RB drawdown to negative pressure, this activity is assumed to be released directly to the environment.
2. MSIV leakage – Leakage through the MSIVs reaches the condenser through the steam lines, except for a small percentage that bypasses the condenser. The condenser is then assumed to leak to the TB and eventually to the MCR and the environment.
3. ESF leakage – This leakage occurs in secondary containment after the DW sprays have been initiated and water from the torus is recirculated back into the DW.
4. Shine from the radioactive cloud in the TB.

5. Shine from other external sources.

The results of the radiological consequences of a LOCA for offsite, TSC, and MCR are as follows:

- Offsite doses – The EAB and LPZ doses for the ground release pathway and elevated release pathway are shown in Table 19.
- MCR doses – The doses to occupants of the MCR are shown in Table 20.
- TSC doses – The dose to an individual inside the TSC is shown in Table 21.

In all cases, the doses are within regulatory limits.

**Table 8. Core Release Rates**

| Group | Elements | Release Fraction | | Release Rate (Frac/hr) | |
|---|---|---|---|---|---|
| | | Gap | Early In-Vessel | Gap | Early In-Vessel |
| Halogens | I, Br | 0.05 | 0.25 | 0.1 | 0.167 |
| Noble Gases | Kr, Xe | 0.05 | 0.95 | 0.1 | 0.633 |
| Alkali Metals | Cs, Rb | 0.05 | 0.20 | 0.1 | 0.133 |
| Tellurium Metals | Sb, Se, Te | 0 | 0.05 | 0 | 0.0333 |
| Barium, Strontium | Ba, Sr | 0 | 0.02 | 0 | 0.0133 |
| Noble Metals | Co, Mo, Pd, Rh, Ru, Tc | 0 | 0.0025 | 0 | 0.00167 |
| Lanthanides | Am, Cm, Eu, La, Nb, Nd, Pm, Pr, Sm, Y, Zr | 0 | 0.0002 | 0 | 0.000133 |
| Cerium Group | Ce, Np, Pu | 0 | 0.0005 | 0 | 0.000333 |
| Release Duration (hr) | | 0.5 | 1.5 | | |

Note: Release rate is obtained by dividing the release fraction by the release duration.

**Table 9. Flow Rates from DW to Torus**

| Time (hr) From | Time (hr) To | Flow rate (cfm) |
|---|---|---|
| 0 | 2.03 | 0 |
| 2.03 | 2.06 | 26457 |
| 2.06 | 2.39 | 685 |
| 2.39 | 3.00 | 349 |
| 3.00 | 720 | 0 |

**Table 10. DW Activity Removal Rates**

| Time (hr) | | $\lambda$ (hr$^{-1}$) |
|---|---|---|
| From | To | |
| 0 | 0.250 | 0.0400 |
| 0.250 | 0.533 | 12.4 |
| 0.533 | 0.595 | 23.7 |
| 0.595 | 2.03 | 16.7 |
| 2.03 | 2.34 | 4.72 |
| 2.34 | 2.67 | 3.69 |
| 2.67 | 3.10 | 2.87 |
| 3.10 | 3.89 | 2.22 |
| 3.89 | 7.38 | 1.67 |
| 7.38 | 720 | 0 |

**Table 11. DW Pressure Used in Containment Activity Removal Model**

| Time Frame (accident hr) | Pressure (psia) |
|---|---|
| 0 – 0.250 | 44.7 |
| 0.250 – 2.033 | 18.1 |
| 2.033 – 24.00 | 25.3 |

**Table 12. DW Temperature Used in Containment Activity Removal Model**

| Time Frame (accident hr) | Temp. (°F) |
|---|---|
| 0 – 0.250 | 343 |
| 0.250 – 2.033 | 152 |
| 2.033 – 24.00 | 92 |

**Table 13. DW Steam Mole Fraction Used in Containment Activity Removal Model**

| Time Frame (accident hr) | Steam Mole Fraction |
|---|---|
| 0 – 0.250 | 0.95 |
| 0.250 – 2.033 | 0.11 |
| 2.033 – 24.00 | 0.02 |

**Table 14. Main Steam Line Deposition Rates**

| Time (hr) | | $\lambda$ ($hr^{-1}$) |
|---|---|---|
| From | To | |
| 0 | 2.56 | 0.669 |
| 2.56 | 3.77 | 0.541 |
| 3.77 | 5.44 | 0.435 |
| 5.44 | 8.05 | 0.346 |
| 8.05 | 12.0 | 0.276 |
| 12.0 | 18.8 | 0.223 |
| 18.8 | 24.1 | 0.199 |
| 24.1 | 720 | 0 |

**Table 15. Condenser Deposition Rates**

| Time (hr) | | $\lambda$ ($hr^{-1}$) |
|---|---|---|
| From | To | |
| 0.0333 | 5.51 | 0.119 |
| 5.51 | 13.8 | 0.0949 |
| 13.8 | 24.0 | 0.0780 |
| 24.0 | 720 | 0.0724 |

**Table 16. Containment Leakage Reduction Parameters**

| | Unit 1 | | | Unit 2 | | |
|---|---|---|---|---|---|---|
| | Post-LOCA Time (hr) | | | Post-LOCA Time (hr) | | |
| | 0 | 24 | 72 | 0 | 24 | 72 |
| DW Pressure (psia) | 65.5 | 25.4 | 20.9 | 62.0 | 25.4 | 20.9 |
| DW Temperature (°F) | 293 | 188 | 160 | 292 | 188 | 160 |
| Flow Ratio | 1.00 | 0.60 | 0.48 | 1.00 | 0.60 | 0.48 |

**Table 17. MSIV Leakage Reduction Parameters**

| | Post-LOCA Time (hr) | | |
|---|---|---|---|
| | 0 | 24 | 72 |
| DW Pressure (psia) | 76.7 | 25.4 | 20.9 |
| DW Temperature (°F) | 340 | 188 | 160 |
| Flow Ratio | 1.00 | 0.59 | 0.47 |

**Table 18. Containment and MSIV Leakage Rates**

| Leakage Source | Leakage Pathway | | Flow Rate (cfm) | | |
|---|---|---|---|---|---|
| | From | To | 2 min | 24 hr | 72 hr |
| Containment Leakage | DW | Condenser or Environment | 0.0244 | 0.0146 | 0.0122 |
| | DW | RB | 1.20 | 0.720 | 0.600 |
| | Torus Air | Condenser or Environment | 0.0183 | 0.0110 | 0.00915 |
| | Torus Air | RB | 0.899 | 0.539 | 0.450 |
| MSIV Leakage | DW | MSL | 0.828 | 0.497 | 0.414 |
| | MSL | Condenser / HPT | 4.38 | 2.63 | 2.19 |
| | MSL | Condenser | 4.36 | 2.62 | 2.18 |
| | MSL | HPT | 0.0219 | 0.0132 | 0.0110 |
| | Condenser | TB or Environment | 2.29 | 1.37 | 1.15 |
| | HPT | TB or Environment | 0.0219 | 0.0132 | 0.0110 |

**Table 19. Offsite Doses for LOCA**

| Release Pathway | Dose (rem TEDE) | | |
|---|---|---|---|
| | EAB | LPZ | Limit |
| Ground | 0.307 | 0.644 | |
| Elevated | 0.033 | 0.110 | |
| Total | 0.34 | 0.75 | 25 |

**Table 20. MCR Doses for LOCA**

| | Dose (rem TEDE) |
|---|---|
| MCR air | 4.32 |
| TB air (external shine) | 0.0024 |
| Ingress/egress (through TB) | 0.57 |
| Other external shine sources | 0.03 |
| Total | 4.9 |
| Limit | 5 |

**Table 21. TSC Dose for LOCA**

| | Dose (rem TEDE) |
|---|---|
| TSC air | 3.9 |
| Limit | 5 |

### 2.5.3 Fuel Handling Accident (FHA)

The radiological dose consequences to occupants of the MCR and the TSC and to persons located at the EAB and LPZ following a postulated FHA are determined using AST assumptions and methodology in accordance with RG 1.183.

Two cases of radioactivity release paths from the RB are evaluated. The first case considers that the SGTS is in operation, so that all releases are filtered and elevated after drawdown of the RB. The second case does not take credit for operation of the SGTS.

2.5.3.1 Inputs and Assumptions

*Iodine Species in Pool*

The iodine released from the fuel into the pool is assumed to be composed of 99.85% elemental and 0.15% organic species, per RG 1.183, Appendix B, Section 2.

*Activity Released*

The FHA is postulated to occur at the earliest possible time of fuel movement following shutdown from full power. The fuel is assumed to decay for 24 hr. The source terms are listed in Table 22.

*Fuel Quantity*

There are 560 fuel bundles in the core. Each GE14 (10x10) fuel bundle contains an average of 87.3 fuel rods.

*Fuel Damage*

The FHA is estimated to result in 172 damaged fuel rods.

*Release Fractions*

As indicated in Appendix B of RG 1.183, all the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that are considered include xenons, kryptons, halogens, cesiums, and rubidiums. The fractions of fission product inventory in the gap are I-131 (0.08), other halogens (0.05), Kr-85 (0.10), other noble gases (0.05), and alkali metals (0.12) (RG 1.183, Table 3). The release fractions are applied in Table 22 to determine the radioactivity released from the damaged fuel.

*Radial Peaking Factor*

The maximum core radial peaking factor of 1.5 is applied to all of the damaged fuel rods.

*Water Depth*

The minimum depth of water above the damaged fuel is 21 ft. Since this is less than the 23 ft depth assumed in the RG 1.183 derivation of the iodine DF, the reduced iodine DF is calculated in this analysis.

*Pool Decontamination Factors*

The retention of noble gases in the water in the spent fuel pool is negligible (i.e., DF of 1). Particulate radionuclides are assumed to be retained by the water in the pool (i.e., infinite DF) (RG 1.183, Appendix B, Section 3).

*Secondary Containment*

The radioactive material that escapes from the pool is released to the environment over a 2-hr time period (RG 1.183, Appendix B, Section 5.3). Two cases of secondary containment isolation are considered. The first case assumes that the secondary containment isolates automatically in the event of a FHA. The second case assumes that isolation does not occur.

Case 1
The time necessary to drawdown the secondary containment (establish a 0.20 in wg negative pressure) is 120 sec. Prior to that time, airborne releases are assumed to be unfiltered at ground level. After drawdown, all of the airborne activity is collected by the SGTS and released. The release is filtered (95% efficient filters for iodines and particulates) and elevated.

Case 2
No credit is taken for secondary containment isolation or operation of the SGTS. The release is assumed to be unfiltered at ground level for the duration of the accident.

*MCR Unfiltered Inleakage*

The unfiltered inleakage for the MCR is conservatively assumed to be 10,000 cfm, which is the same unfiltered inleakage value assumed for the TSC.

*Accident Duration*

MCR and TSC doses are calculated for a 30-day period, since radioactivity that is brought into those rooms during the first 2 hr of the accident will continue to expose occupants until it is removed by air transfer or decay. Offsite doses are calculated for a 2-hr period, since the radioactive cloud is assumed to move past these locations during this time interval and no further exposure occurs (RG 1.183, Table 6).

*Atmospheric Dispersion*

MCR, TSC, and offsite $\chi$/Q values are listed in Table 4 and Table 5. The MCR values are applied at the TSC since the MCR values are bounding.

2.5.3.2 Method of Evaluation

Two cases of radioactivity release paths from the RB are evaluated. The first case considers that the SGTS is in operation, so that all releases are filtered and elevated after drawdown of the RB. The second case does not take credit for the SGTS.

*Decontamination Factor*

RG 1.183 allows the use of an overall DF of 200 if the depth of water over the damaged fuel is 23 ft or greater (Appendix B, Section 2). If the depth of water is not 23 ft, the DF is determined on a case-by-case method.

For HNP, the minimum requirement for water level above the spent fuel is 21 ft. Thus, the DF as a function of water depth is derived so that an appropriately conservative value is applied to the iodine activity released from the fuel.

The overall effective DF represents a composite for the different iodine species. RG 1.183 stipulates an iodine species split of 99.85% elemental (inorganic) iodine and 0.15% organic iodine. The DF for organic iodine is 1, which means that organic iodine is not retained in water.

Using the DF values from RG 1.183 of 500 for inorganic iodine and 1 for organic iodine, the overall effective DF is 286. Since RG 1.183 allows the use of an overall DF of 200, the factor of conservatism is therefore 286/200, or 1.43. This factor is applied to the effective DFs calculated as a function of water depth and is shown in Table 23. The last column shows the adjusted DF values.

For the FHA analysis, an overall effective DF of 142 (corresponding to 21 ft of water above the damaged fuel) is used.

2.5.3.3 Results

The doses to MCR and TSC occupants and to persons at the EAB and LPZ are listed in Table 24. All doses are below the respective acceptance criteria.

Based on the above results, the following conclusions may be drawn:

- The MCR and TSC can tolerate significant unfiltered inleakage during a FHA.
- The SGTS is not required to meet the dose criteria following a postulated fuel handling accident.

**Table 22. Activity Releases from FHA**

| | Element | Isotope | Core Source Term at t=1 day (Ci/MWt)[1] | Gap Fraction[2] | Release from Fuel (Ci)[3] | Water DF | Release from Water (Ci)[4] |
|---|---|---|---|---|---|---|---|
| Noble Gases | Xenon | Xe-133 | 5.10E+04 | 0.05 | 4.17E+04 | 1 | 4.17E+04 |
| | | Xe-135 | 1.41E+04 | 0.05 | 1.15E+04 | 1 | 1.15E+04 |
| | | Xe-137 | 0.00E+00 | 0.05 | 0.00E+00 | 1 | 0.00E+00 |
| | | Xe-138 | 0.00E+00 | 0.05 | 0.00E+00 | 1 | 0.00E+00 |
| | | Xe-131m | 3.02E+02 | 0.05 | 2.47E+02 | 1 | 2.47E+02 |
| | | Xe-133m | 1.45E+03 | 0.05 | 1.19E+03 | 1 | 1.19E+03 |
| | | Xe-135m | 6.68E+02 | 0.05 | 5.46E+02 | 1 | 5.46E+02 |
| | Krypton | Kr-85 | 3.78E+02 | 0.1 | 6.18E+02 | 1 | 6.18E+02 |
| | | Kr-87 | 2.78E-02 | 0.05 | 2.27E-02 | 1 | 2.27E-02 |
| | | Kr-88 | 5.31E+01 | 0.05 | 4.34E+01 | 1 | 4.34E+01 |
| | | Kr-89 | 0.00E+00 | 0.05 | 0.00E+00 | 1 | 0.00E+00 |
| | | Kr-83m | 1.31E+01 | 0.05 | 1.07E+01 | 1 | 1.07E+01 |
| | | Kr-85m | 1.71E+02 | 0.05 | 1.40E+02 | 1 | 1.40E+02 |
| Halogens | Iodine | I-129 | 1.23E-03 | 0.05 | 1.01E-03 | 142 | 7.08E-06 |
| | | I-130 | 2.73E+02 | 0.05 | 2.23E+02 | 142 | 1.57E+00 |
| | | I-131 | 2.52E+04 | 0.08 | 3.30E+04 | 142 | 2.32E+02 |
| | | I-132 | 3.21E+04 | 0.05 | 2.63E+04 | 142 | 1.85E+02 |
| | | I-133 | 2.54E+04 | 0.05 | 2.08E+04 | 142 | 1.46E+02 |
| | | I-134 | 1.35E-03 | 0.05 | 1.10E-03 | 142 | 7.78E-06 |
| | | I-135 | 4.17E+03 | 0.05 | 3.41E+03 | 142 | 2.40E+01 |
| | | I-136 | 0.00E+00 | 0.05 | 0.00E+00 | 142 | 0.00E+00 |
| | | I-137 | 0.00E+00 | 0.05 | 0.00E+00 | 142 | 0.00E+00 |
| | | I-138 | 0.00E+00 | 0.05 | 0.00E+00 | 142 | 0.00E+00 |
| | Bromine | Br-82 | 1.13E+02 | 0.05 | 9.24E+01 | 142 | 6.51E-01 |
| | | Br-83 | 3.37E+00 | 0.05 | 2.76E+00 | 142 | 1.94E-02 |
| | | Br-84 | 1.48E-10 | 0.05 | 1.21E-10 | 142 | 8.53E-13 |
| | | Br-85 | 0.00E+00 | 0.05 | 0.00E+00 | 142 | 0.00E+00 |
| | | Br-87 | 0.00E+00 | 0.05 | 0.00E+00 | 142 | 0.00E+00 |
| | | Br-88 | 0.00E+00 | 0.05 | 0.00E+00 | 142 | 0.00E+00 |
| Alkali Metals | Cesium | Cs-134 | 6.82E+03 | 0.12 | 1.34E+04 | Infinite | 0.00E+00 |
| | | Cs-135 | 2.36E-02 | 0.12 | 4.63E-02 | Infinite | 0.00E+00 |
| | | Cs-136 | 2.07E+03 | 0.12 | 4.06E+03 | Infinite | 0.00E+00 |
| | | Cs-137 | 4.14E+03 | 0.12 | 8.13E+03 | Infinite | 0.00E+00 |
| | | Cs-138 | 2.97E-09 | 0.12 | 5.82E-09 | Infinite | 0.00E+00 |
| | | Cs-139 | 0.00E+00 | 0.12 | 0.00E+00 | Infinite | 0.00E+00 |
| | | Cs-134m | 5.33E+00 | 0.12 | 1.05E+01 | Infinite | 0.00E+00 |
| | Rubidium | Rb-86 | 6.80E+01 | 0.12 | 1.33E+02 | Infinite | 0.00E+00 |
| | | Rb-87 | 0.00E+00 | 0.12 | 0.00E+00 | Infinite | 0.00E+00 |
| | | Rb-88 | 5.93E+01 | 0.12 | 1.16E+02 | Infinite | 0.00E+00 |
| | | Rb-89 | 0.00E+00 | 0.12 | 0.00E+00 | Infinite | 0.00E+00 |
| | | Rb-90 | 0.00E+00 | 0.12 | 0.00E+00 | Infinite | 0.00E+00 |
| Total | | | 1.69E+05 | | 1.66E+05 | | 5.66E+04 |

Notes:
1. Core inventory at t=1 day
2. Gap fractions from RG 1.183, Table 3
3. Release from fuel is core inventory times power level times fraction of core damaged times radial peaking factor times gap fraction times 10% margin
4. Release from water is release from fuel divided by water DF

**Table 23. Decontamination Factors as a Function of Water Depth**

| Water Depth (ft) | DF Inorganic | DF Organic | DF Effective | Adjusted DF |
|---|---|---|---|---|
| 23 | 500 | 1 | 286 | 200 |
| 22.5 | 437 | 1 | 264 | 185 |
| 22 | 382 | 1 | 243 | 170 |
| 21.5 | 333 | 1 | 222 | 156 |
| 21 | 291 | 1 | 203 | 142 |
| 20.5 | 254 | 1 | 184 | 129 |

**Table 24. Doses from FHA**

| Location | Dose (rem TEDE) | | |
|---|---|---|---|
| | Case 1: SGTS | Case 2: No SGTS | Limit |
| MCR | 0.72 | 3.5 | 5.0 |
| TSC | 0.80 | 3.9 | 5.0 |
| EAB | 0.25 | 1.2 | 6.3 |
| LPZ | 0.25 | 1.2 | 6.3 |

## 2.5.4 Control Rod Drop Accident (CRDA)

The radiological dose consequences to occupants of the MCR and the TSC and to persons located at the EAB and LPZ following a postulated CRDA are determined using AST assumptions and methodology in accordance with RG 1.183.

Conservative estimates for the source of unfiltered inleakage are assumed for each dose receptor. For the MCR, the inleakage is assumed to come from the TB. Ingress and egress doses to MCR operators passing through the TB are included in the total dose. For the TSC, inleakage is assumed to come from the environment.

### 2.5.4.1 Inputs and Assumptions

*Fuel Quantity*

There are 560 fuel bundles in the core. Each GE14 (10x10) fuel bundle contains an average of 87.3 fuel rods.

*Fuel Damage*

The control rod drop is estimated to result in 1,200 damaged fuel rods. Of those, 1,189 rods experience cladding failure and 11 rods experience melting.

Cladding Failure:
As indicated in Appendix C of RG 1.183, the activity in the gap of the fuel rods is assumed to be 10% of the core inventory for noble gases and iodines. In addition, Table 3 of RG 1.183 lists fission product inventories in the gap of other groups of elements for non-LOCA events. Although it is not specifically stated that these additional elements should be included in the CRDA, Appendix C of RG 1.183 refers to "remaining radionuclides" and "particulate radionuclides." For completeness, these additional elements are conservatively included in the radiological analysis. Thus, 5% of halogens (other than iodine) and 12% of alkali metals (cesium and rubidium) are also assumed to be in the gap of the fuel rods. All of the gap activity of the 1,189 breached fuel rods is assumed to be released as a result of the accident.

Melting:
As indicated in Appendix C of RG 1.183, 100% of the noble gases and 50% of the iodines contained in the 11 fuel rods that melt are released to the reactor coolant. In addition, Table 1 of RG 1.183 lists fission product release fractions for other groups of elements for design basis LOCA events. Although it is not specifically stated that these activities should be included in the CRDA, Appendix C of RG 1.183 refers to "remaining radionuclides" and "particulate radionuclides." For completeness, these additional elements are conservatively included in the radiological analysis for the rods that melt. Therefore, 30% of halogens (other than iodines), 25% of alkali metals (cesium and rubidium), 5% of tellurium, 2% of barium and strontium, 0.25% of noble metals, 0.05% of cerium, and 0.02% of lanthanides are also assumed to be released from the 11 melted fuel rods.

*Radial Peaking Factor*

The maximum core radial peaking factor of 1.5 is applied to all of the damaged fuel rods.

*Mixing in Vessel*

All of the activity released from the fuel is instantaneously mixed in the reactor coolant within the pressure vessel (RG 1.183, Appendix C, Section 3.1).

*Removal in Vessel*

No credit is taken for partitioning in the pressure vessel or for removal by the steam separators (RG 1.183, Appendix C, Section 3.2).

*Vessel Release Percentages*

Of the activity released from the reactor coolant within the pressure vessel, 100% of the noble gases, 10% of the iodine, and 1% of the remaining radionuclides are assumed to reach the turbine and condensers (RG 1.183, Appendix C, Section 3.3).

*Turbine/Condenser Release Percentages*

Of the activity that reaches the turbine and condenser, 100% of the noble gases, 10% of the iodine, and 1% of the particulate radionuclides are available for release to the environment (RG 1.183, Appendix C, Section 3.4).

*Iodine Species of Release*

The species of iodine released from the turbine/condenser is assumed to be 97% elemental and 3% organic (RG 1.183, Appendix C, Section 3.6). It is noted that RG 1.183, Appendix C, Section 3.6 also states that the iodine species released from the reactor coolant within the pressure vessel should be assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic. However, in this analysis no iodine species specific mechanisms act upon the iodine prior to release from the turbine/condenser so the iodine species percentages are applied at the time of release from the fuel.

*Turbine/Condenser Leak Rate*

The turbine/condenser leaks at a rate of 1% per day for a period of 24 hr, at which time the leakage is assumed to terminate (RG 1.183, Appendix C, Section 3.4).

*Mechanical Vacuum Pump*

For offsite and TSC doses, a second release path is evaluated to address the possibility of a forced flow path from the turbine or condenser (RG 1.183, Appendix C, Section 3.4, footnote 2). The mechanical vacuum pump normally discharges to the plant stack through the gland-seal holdup line, which provides holdup for up to 2 min, but no filtration. The pump trips on high MSL radiation, but the release is conservatively assumed to continue for 24 hr. The mechanical vacuum pump flow is assumed to be 2,200 cfm for 24 hr, at which time the release is assumed to terminate.

*Accident Duration*

MCR and TSC doses are calculated for a 30-day period, since radioactivity that is brought into those rooms during the first 24 hr of the accident will continue to expose occupants until it is removed by air transfer or decay. Also, unfiltered inleakage to the MCR from the contaminated TB is assumed to continue for the 30-day accident duration. EAB doses are calculated for a 2-hr period. LPZ doses are calculated for a 24-hr period, since the radioactive cloud is assumed to move past these locations during this time interval and no further exposure occurs.

*Atmospheric Dispersion*

Design basis $\chi/Q$ values for the MCR, TSC, and offsite are listed in Table 4 and Table 5. For MCR, the release point is assumed to be the RB vent. For the EAB and LPZ, ground level release is assumed. The MCR values are also applied at the TSC since the MCR values are bounding. Since the release terminates at 24 hr, $\chi/Q$ values are not used at the LPZ after that time.

For the case of forced flow from the turbine/condenser, the release is elevated through the main stack.

*Dose in MCR from External Sources*

Since the MCR walls are 2 ft concrete and the roof is 2.5 ft concrete, the dose due to external airborne activity is assumed to be negligible compared to the dose received from activity within the MCR. The LOCA analysis demonstrates that the MCR dose due to TB activity is negligible compared to that due to activity within the MCR.

2.5.4.2 Method of Evaluation

Doses are calculated using the guidance provided in the main body and Appendix C of RG 1.183. LocaDose is used to calculate the dose in the MCR and TSC and at the EAB and LPZ. The two

LocaDose modules utilized in this analysis are Activity Transport Program and Dose Calculation Program. A description of LocaDose modules is given in Section 2.2.1.

*Transport of Radioactivity and Doses*

Two models for the transport of radioactivity from the turbine/condenser to the MCR and to the environment are considered, to conservatively maximize the dose to each.

The first model considers holdup of radioactivity in the TB so that MCR doses are calculated conservatively, since the MCR is located in the TB. The TB exhausts to the environment starting at 9 hr with a flow of 15,000 cfm via the RB vent. Ingress and egress doses are also calculated for MCR operators who pass through the TB to enter and exit the MCR.

The second model does not consider holdup of radioactivity within the TB so that doses to occupants of the TSC and persons at the EAB and LPZ are calculated conservatively. Leakage from the turbine/condenser is released directly to the environment at ground level.

*MCR Ingress/Egress Doses*

The total dose to MCR operators is the sum of the dose received while occupying the MCR for 30 days plus the dose received while traveling to and from the MCR through the TB. The methodology for calculating ingress/egress dose is the same as that used in the dose calculations for the LOCA. This methodology is explained in Section 2.5.2.2.

2.5.4.3 Results

Table 25 shows that the doses to persons at the EAB and the LPZ are below the acceptance criteria for a postulated CRDA. Table 26 shows that the dose to MCR personnel is within the regulatory limit. Table 27 shows that the TSC dose is well within the acceptance criterion.

**Table 25. Offsite Doses from CRDA**

| | Dose (rem TEDE) | | |
|---|---|---|---|
| | EAB | LPZ | Limit |
| Design Basis | 0.047 | 0.094 | 6.3 |
| Forced Flow | 0.333 | 0.540 | 6.3 |

**Table 26. MCR Doses from CRDA**

| | Dose (rem TEDE) |
|---|---|
| MCR air | 3.61 |
| Ingress / Egress | 0.23 |
| Total | 3.8 |
| Regulatory Limit | 5 |

**Table 27. TSC Doses from CRDA**

| | Dose (rem TEDE) | |
|---|---|---|
| | TSC | Limit |
| Design Basis | 0.32 | 5 |
| Forced Flow | 0.81 | 5 |

## 2.5.5 Main Steam Line Break (MSLB)

This analysis calculates the radiological dose consequences to occupants of the MCR and the TSC and to persons at the EAB and LPZ following a postulated MSLB using AST assumptions and methodology in accordance with RG 1.183.

A maximum primary coolant iodine concentration of 2.0 μCi/g DE I-131, which reflects a proposed change in the applicable TS, is assumed.

### 2.5.5.1 Inputs and Assumptions

*Fuel Damage*

The temperature and pressure transients resulting from this event are not severe enough to cause fuel damage.

*Primary Coolant Iodine Activity during Accident*

For the case of a pre-accident spike, the iodine concentration in the primary coolant is assumed to be 2.0 μCi/g DE I-131. For the case of maximum equilibrium value, the iodine concentration in the primary coolant is assumed to be 0.2 μCi/g DE I-131, the value allowed by the TSs for continued full power operation.

*Primary Coolant Iodine Activities during Normal Operation*

During normal operation, the relative distribution of iodine isotopes in the primary coolant is given in Table 28.

*Noble Gas Release Rate*

The noble gas release rates from the core are shown in Table 29 The values in Table 29 correspond to a decayed total release rate of 0.3 Ci/sec after a delay of 30 min for the offgas system. The TS limit for the delayed offgas release rate is 0.240 Ci/sec. Hence, the noble gas release rates in Table 29 bound the value in the TS. The noble gas release rates from the break are assumed to be the same as from the core as shown in Table 29, assuming no decay.

*MSIV Closure Time*

The high-flow signal from the break is assumed to initiate MSIV closure within 0.5 sec of the break. The maximum time required to isolate the MSIV is 5 sec. Therefore, total time from the break to MSIV isolation is assumed to be 5.5 sec.

*Fluid Release from Reactor Coolant*

The steam that is released expands due to the lower atmospheric pressure and becomes superheated. The initial mass of the saturated liquid that is released equals the mass of saturated liquid and vapor at atmospheric pressure (14.7 psia) and 212°F. Enthalpy is constant through the phase change as part of the released saturated liquid flashes to vapor.

*Steam Blowdown*

The blowdown rate through the break is given in Table 30.

*Mixture Quality*

A mixture quality of 7% is assumed for the mixture portion of the blowdown.

*Turbine Building*

In calculating the MCR dose, it is assumed that the activity released from the break is uniformly mixed within the total TB free volume. This is reasonable given the force with which the steam is released from the break.

*Dose in MCR from External Sources*

Since the MCR walls are 2 ft concrete and the roof is 2.5 ft concrete, the dose due to external airborne activity is assumed to be negligible compared to the dose received from activity within the MCR. The LOCA analysis demonstrates that the MCR dose due to TB activity is negligible compared to that due to activity within the MCR.

*Atmospheric Dispersion*

MCR atmospheric dispersion factors ($\chi/Q$) for a ground level release through the RB vent are shown in Table 4. The MCR values are applied to the TSC since the MCR values are bounding. The EAB and LPZ $\chi/Q$ values for a ground level release are shown in Table 5.

*Accident Duration*

The release from the break is assumed to be an instantaneous puff. MCR, TSC, and LPZ doses are calculated assuming an exposure time of 30 days. The EAB dose is calculated over a 2-hr period which yields the maximum dose which, for a puff release, is the first 2 hr.

2.5.5.2 Method of Evaluation

Doses are calculated using the Activity Transport Program and Dose Calculation Program modules of LocaDose. The following sequence of events is assumed to occur:

1. Before the accident, the reactor is assumed to be in hot standby mode to maximize the inventory lost through the break prior to isolation.
2. One of the MSLs outside the RB is completely severed, releasing steam.
3. Within 0.5 sec, a high-flow signal initiates MSIV closure.
4. Rapid depressurization of the reactor pressure vessel causes the water level to rise, releasing a steam-water mixture from the break.
5. The reactor scrams.
6. Within the maximum time allowed by the TSs, the MSIVs are fully closed, terminating the release.
7. The total mass of coolant released is that amount in the steam line and connecting lines at the time of the break plus the amount that passes through the valves prior to closure.
8. The released steam forms a large cloud.

The mass released includes the steam originally in the line as well as from a portion of the saturated liquid which spills from the break and flashes to steam. The flashing fraction of the released liquid assumes a constant enthalpy process.

*Offsite Dose Analysis*

For the dose calculations for persons at the EAB and LPZ, it is assumed that the accident damages the TB such that it is not able to contain the steam that is released from the break. The steam is released to the environment as a puff, resulting in offsite doses. Two scenarios are evaluated. In the first case, the primary coolant iodine concentration corresponds to a pre-accident spike. In the second case, the primary coolant iodine concentration corresponds to an equilibrium value.

*MCR Dose Analysis*

For doses to occupants of the MCR, it is assumed that the TB contains the steam that is released from the break. This contained source is available for direct leakage into the MCR. The primary coolant iodine concentration corresponds to a pre-accident spike, which bounds the equilibrium value case.

*TSC Dose Analysis*

For doses to occupants of the TSC, the model used is the same as that used for calculating EAB and LPZ doses. The primary coolant iodine concentration corresponds to a pre-accident spike, which bounds the equilibrium value case.

*MCR Ingress/Egress*

MCR ingress/egress doses apply to MCR operators as they walk across the TB deck when entering and leaving the MCR. These doses are calculated by determining the average TB dose rate during a time interval and multiplying by the exposure duration. The methodology for calculating ingress/egress dose is the same as that used in the dose calculations for the LOCA. This methodology is explained in Section 2.5.2.2.

*Impact of Cesium*

An evaluation is performed to determine the impact of cesium released during the MSLB on MCR and offsite doses. The following cesium isotopes are considered: Cs-134, Cs-136, Cs-137, and Cs-138. For MCR doses, the cesium dose contribution represents a very small fraction of the dose received from iodine and noble gas isotopes. This difference is considered to be within the accuracy of the MSLB calculation. For offsite doses, the cesium contribution represents a relatively larger fraction of the total dose since offsite releases do not have the benefit of particulate filters; however, the total offsite dose, including the cesium dose contribution, is less than one percent of the regulatory limit. It is therefore concluded that cesium has an insignificant impact on both MCR and offsite doses.

2.5.5.3 Results

The EAB and LPZ doses are shown in Table 31 for both cases analyzed. The MCR doses are shown in Table 32. The TSC doses are shown in Table 33. For all cases analyzed, doses are within the regulatory limits.

**Table 28. Primary Coolant Iodine Activities**

| Isotope | Primary Coolant Activity (μCi/g) |
|---|---|
| I-131 | 0.018 |
| I-132 | 0.16 |
| I-133 | 0.12 |
| I-134 | 0.31 |
| I-135 | 0.17 |
| Total | 0.778 |

**Table 29. Noble Gas Release Rates**

| Isotope | Release Rate (Ci/sec) |
|---|---|
| Kr-83m | 1.02E-02 |
| Kr-85m | 1.83E-02 |
| Kr-85 | 6.00E-05 |
| Kr-87 | 6.00E-02 |
| Kr-88 | 6.00E-02 |
| Kr-89 | 3.90E-01 |
| Xe-131m | 4.50E-05 |
| Xe-133m | 8.70E-04 |
| Xe-133 | 2.46E-02 |
| Xe-135m | 7.80E-02 |
| Xe-135 | 6.60E-02 |
| Xe-137 | 4.50E-01 |
| Xe-138 | 2.67E-01 |
| Total | 1.43E+00 |

**Table 30. Steam Blowdown Rate**

| Time After Break (sec) | Mass Flow (lbm/sec) | Enthalpy (Btu/lbm) |
|---|---|---|
| 0 | 5,300 | 1,191.5 |
| 2.75 | 4,500 | 1,191.5 |
| 2.76 | 19,600 | 589.3 |
| 4.0 | 19,500 | 589.5 |
| 5.5 | 0 | 589.5 |

**Table 31. Offsite Doses from MSLB**

| Scenario | DE I-131 Activity (μCi/g) | Dose (rem TEDE) | | |
|---|---|---|---|---|
| | | EAB | LPZ | Limit |
| Pre-accident spike | 2.0 | 0.15 | 0.15 | 25 |
| Equilibrium iodine activity | 0.2 | 0.015 | 0.015 | 2.5 |

**Table 32. MCR Doses from MSLB**

| | Dose (rem TEDE) |
|---|---|
| MCR air | 3.70 |
| Ingress / Egress | 0.22 |
| Total | 3.9 |
| Regulatory limit | 5 |

**Table 33. TSC Doses from MSLB**

| | Dose (rem TEDE) |
|---|---|
| TSC air | 0.43 |
| Regulatory limit | 5 |

## 2.6 Suppression Pool pH Control

Suppression pool water will retain soluble gases and soluble fission products such as iodine and cesium. Once deposited, the iodine will remain in solution as long as the suppression pool pH is maintained at or above 7 (RG 1.183, Appendix A, Section 2). The pH of the suppression pool water is calculated as a function of time to demonstrate that the pH remains at or above 7 for the duration of the DBA LOCA.

### 2.6.1 Inputs and Assumptions

*Core Fission Product Inventory*

For radiolysis of water and the production of nitric acid, the fission product and actinide decay power is assumed to be a function of fission product/actinide mass grouped into eight categories per Table 3.4 of Reference 4. This power specification is inherently conservative since it was developed for relatively low burnup, and as burnup increases, the power per unit mass of fission products/actinides decreases. The group fission product/actinide mass inventories used for HNP are based on values for BWRs of similar thermal power with relatively high burnup (making the power specification very conservative), including a multiplication factor of 1.1 for additional conservatism.

For radiolysis of cable and the production of hydrochloric acid, the power specification is based on a conservative activity inventory and associated gamma and beta MeV/sec/MWt that is generic to the STARpH methodology (see Section 2.6.2).

*Aerosol Fraction*

The fraction of aerosol in the source term in the suppression pool is 0.90. Given that the spray will tend to wash any aerosol which deposits on elevated surfaces into the sump, the actual fraction of aerosol in the water pool is expected to be essentially 100%. Thus use of 90% is conservative since it will overestimate the radiation level in the DW vapor space and thus overestimate the concentration of hydrochloric acid, [HCl], from radiolysis of chloride-bearing cable insulation.

*Organic Acid from Paints*

Organic acid from paints can be neglected. The hydrogen ion concentration, [H+], from the production of organic acid in the suppression pool is expected to be a small fraction of the total [H+] from the nitric and hydrochloric acid calculated to be produced by the Radiolysis of Water and Radiolysis of Cable models of the STARpH code.

*Sodium Pentaborate Addition*

The SLC system is actuated and the SPB solution is injected into the pool within several hours of accident initiation. A core damage event large enough to release the substantial quantities of fission products in the time frame considered for the AST in RG 1.183 will be very evident to the operators (e.g., core outlet temperature, radiation level in the DW, pressure and temperature in the DW, hydrogen level in the DW) within minutes of the initiating event. Thus it is reasonable to assume the HNP emergency operating procedures provide for SLC system actuation within approximately 2 hr of accident initiation.

If SLC injection is into the pool (i.e., into the reactor vessel with the vessel communicating with the pool as in a recirculation line break), significant mixing will occur quickly, on the order of 1 hr based on an RHR flow rate of about 10,000 gpm and pool volume of 7E+05 gallons.

If the reactor vessel is not immediately communicating with the pool, an additional few hours is assumed to assure communication with the pool or inject SPB to the pool via an alternate pathway.

*Unbuffered Pool pH*

The unbuffered pH of the pool should remain above 7 for at least several hours. The acid added from radiolysis of water (HNO3) and radiolysis of cable (HCl) is not enough to neutralize the hydroxyl ion concentration, [OH-], from fission product cesium until approximately 1 day after accident initiation.

*Pool Temperature*

The average temperature of the pool over 30 days is 155°F. The dissociation constant and starting pH are somewhat, but not strongly, temperature dependent. The average temperature of the pool over 30 days is calculated to be 155°F for a BWR recently studied that operates at a 23% higher thermal power than the HNP units, but has a pool volume about 40% greater than the HNP units.

The design inputs to the pH calculation are given in Table 34.

### 2.6.2 Method of Evaluation

The BWR version of the Radiolysis in Water model in the STARpH code calculates the hydroxyl ion concentration from fission product cesium, and nitric acid concentration in the containment water pool generated by radiolysis. The Radiolysis of Cable model in the STARpH code calculates the hydrochloric acid concentration as a result of radiolysis of electrical cable insulation. From these two calculations, the net hydrogen ion concentration added to the pool is calculated over time.

A calculation is performed to determine the amount of SPB buffer added to the pool from the SLC system. From this, the concentration of boron in the water pool is determined.

The Add Acid model of STARpH is used to determine pH as a function of time using the [$H^+$] added, the concentration of Boron in the pool, the boron buffer dissociation constant, and the starting pH of the buffer solution.

The reliability of the SLC system to perform the post-LOCA function (injection of pH buffering agent) is discussed in Section 2.7.2.

### 2.6.3 Results

The boron buffering is conservatively assumed to begin at 5 hr. Thus for times up to 5 hr, the pH is determined by the net [$OH^-$] resulting from the initial pH, HI, CsOH, and $HNO_3$ and the [$H^+$] added to the pool from [HCl]. For time points 1 hr and 2 hr, pH is indicated simply as > 8.0 on the basis of [$OH^-$] from fission product cesium. From 5 hr on, the effect of cesium is neglected and pH is obtained by applying the addition of [$H^+$]. The results are shown in Table 35.

It is determined that the calculated required quantity of SPB is met by the current TS limit. The pH of the containment water pool for the DBA LOCA is 7.7, or above, over a period of 30 days following accident initiation.

**Table 34. Design Inputs for pH Calculation**

| Input / Assumption | Value |
|---|---|
| Suppression Pool Volume (minimum/maximum) | 85,110 $ft^3$ / 89,670 $ft^3$ |
| RCS Inventory (volume in vessel and recirculation loops) | 9,965 $ft^3$, 18,000 lbm steam |
| Initial Suppression Pool pH | 7.2 |
| Electrical Cable Insulation (Hypalon) Mass (Unit1 / Unit 2) | 6,859 lbm / 4,215 lbm |
| Fraction of Cable with Chloride-bearing Insulation in Conduit | 10% |
| Unit 1 DW Free Volume | 146,010 $ft^3$ |
| Unit 1 Minimum Pressure Suppression Chamber Free Volume | 112,900 $ft^3$ |
| Unit 2 DW Free Volume | 146,266 $ft^3$ |
| Unit 2 Minimum Pressure Suppression Chamber Free Volume | 109,800 $ft^3$ |
| Mass of SPB Available for Injection | 1,975 lbm |
| Boron Enrichment | 60% $B_{10}$ |

**Table 35. Suppression Pool pH vs. Time**

| Time | pH | |
|---|---|---|
| | Unit 1 | Unit 2 |
| 1 hr | >8.0 | >8.0 |
| 2 hr | >8.0 | >8.0 |
| 5 hr | 8.4 | 8.4 |
| 12 hr | 8.3 | 8.4 |
| 1 day | 8.3 | 8.3 |
| 3 day | 8.2 | 8.2 |
| 10 day | 7.9 | 8.1 |
| 20 day | 7.8 | 8.0 |
| 30 day | 7.7 | 7.9 |

## 2.7 Crediting of Non-Safety Related Systems

The DBA analyses for the LOCA, FHA, CRDA, and MSLB use inputs that rely on the availability of many plant systems in order to mitigate the effects of the accidents. For the LOCA, CRDA, and MSLB, the analyses require the operation and structural integrity of a small number of systems that, although demonstrated to be highly reliable, are not safety related. These systems (including the applicable DBAs that credit them in the analyses) are:

- MSIV ALT Pathway (LOCA)
- SLC System (LOCA)
- TB Ventilation System (LOCA, CRDA, MSLB)

This section demonstrates the reliability of these systems in the various radiological dose consequence analyses, along with NRC-approved methodologies where applicable.

### 2.7.1 MSIV Alternate Leakage Treatment

RG 1.183, Appendix A, Section 6, allows credit for reduction in MSIV releases due to holdup and deposition in the main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake. Per RG 1.183, an acceptable model for evaluating reduction of MSIV releases is provided in General Electric Topical Report NEDC-31858P-A, "BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems" (Reference 3).

The NRC Safety Evaluation for NEDC-31858P-A (Reference 5) identified limitations to be addressed as part of a plant specific application of the ALT methodology. These limitations relate to assuring that the ALT pathway for MSIV leakage is functionally reliable commensurate with its intended safety function and assuring the pathway, including the main condenser, is seismically rugged.

The use of MSIV ALT was previously approved for use on Unit 2 (Reference 1). The following discussion is with regard to Unit 1.

2.7.1.1 ALT Pathway Description

The HNP ALT pathway for Unit 1 utilizes the large volume of the MSLs and the main condenser to provide holdup and plate-out of fission products that may leak through the closed MSIVs. The primary components of the ALT pathway are the main condenser, the MSLs from the MSIVs to the turbine stop and bypass valves, and the drain piping which originates downstream of the outboard MSIVs and terminates at the main condenser. The condenser forms the ultimate boundary of the ALT pathway. Existing valves upstream of the condenser are used to establish the flow path and isolate the boundaries of the path and to limit the extent of seismic verification walkdown. The model for evaluating reduction of MSIV leakage is provided in Reference 3. Figure 1 shows a schematic of the primary and secondary ALT pathways.

The ALT pathway utilizes MSL drains to direct MSIV leakage to the main condenser. The ALT path is from the downstream side of the MSIVs through four 2-in lines which join a 3-in drain line to the main condenser. The path to the condenser is through motor operated valves (MOVs) 1B21-F020 and 1B21-F021. MOV 1B21-F020 is normally open and will remain open. MOV 1B21-F021 is normally closed and must be opened. Class 1E power is supplied to 1B21-F021 to assure the ability to open the valve. It will be opened by operator action from the MCR to initiate the flow path to the condenser. Valve 1B21-F019 is a normally closed MOV in the drain line upstream of 1B21-F020 and 1B21-F021. It is a primary containment isolation valve and will close or remain closed to maintain the upstream boundary. MOV 1B21-F038, a 2-in drain valve located upstream of 1B21-F021 and downstream of 1B21-F019, is normally closed and will remain closed. As the flow path is via a 3-in line without an orifice, even in the case of loss of offsite power (LOSP), the drain path to the condenser is open and would be available.

For additional assurance that the ALT pathway boundary is isolated and the release is via the condenser, automatic and operator actions will be taken to close boundary valves downstream of the MSIVs and upstream of the condenser. In the event of a LOCA, the MSIVs, the turbine stop valves and the turbine bypass valves will automatically close. The reactor feed pump turbine stop valves, 1N11-F177 and 1N11-F178, which are hydraulically operated, close on an automatic or manual trip of the reactor feed pump turbines.

Operator action is required to isolate steam to the second stage moisture separator reheaters by closing MOVs 1N38-F101A and 1N38-F101B from the MCR. Steam to the steam jet air ejectors will be isolated by closing MOVs 1N11-F001A and 1N11-F001B at local panel 1H21-P216 and manual drain valves 1N11-F039 and 1N11-F041 locally. The seal steam line that comes off of MSL "C" will be isolated by closing MOVs 1N33-F012 and 1N33-F013 from the MCR. Two manual drain valves, 1N11-F043 and 1N11-F044, on the steam lines to the reactor feed pump turbines, will be closed locally by operators.

The ALT pathway must be capable of performing its post-LOCA function during and following a DBE, assuming offsite power is not available. The valves required to be opened in order to establish the path to the condenser, and boundary valves required to be closed to establish the path boundary, are included in the plant's Inservice Inspection Program. The only active valve required to open to establish the ALT pathway, MOV 1B21-F021, is powered from Class 1E power sources and can be opened from the MCR in the event of LOSP.

In the unlikely event that 1B21-F021 fails to open, a secondary passive path with an orifice also exists. In this case, part of the flow would go through a normally open bypass with a 0.103 in diameter orifice around 1B21-F021. The remainder would go to the condenser via the main steam stop and control valves before seat drain lines which contain a 0.850 in diameter restricting orifice.

#### 2.7.1.2 ALT Boundary Seismic Evaluation

The primary ALT boundary components relied upon for pressure boundary integrity are: (1) the main condenser, (2) the MSLs from the MSIVs to the turbine stop and bypass valves, and (3) the main steam turbine bypass and drain line piping to the condenser. The BWR Owners' Group (BWROG) has performed a comprehensive evaluation of the capability of similar components in actual earthquakes. Based on this evaluation of earthquake experience data, the BWROG has developed an approach of verifying the seismic adequacy of the leakage path which is based on utilizing the earthquake experience-based methodology, supplemented by a plant-specific walkdown and analytical evaluations. This methodology is provided in General Electric proprietary report NEDC-31858P. In 1999, the NRC issued a Safety Evaluation Report (SER) (Reference 5) on the GE report. In this SER, the NRC staff stated it considers the BWROG report acceptable for use in individual plant submittals on MSIV leakage issues, subject to the conditions and limitations described in the SER.

A review and evaluation were performed for HNP Unit 1 following the BWROG methodology with appropriate consideration of the conditions and limitations of the NRC SER. Seismic hazard issues identified during the review and evaluation were identified and corrective actions specified. The results of the review and evaluation demonstrate, with the incorporation of the corrective actions, that the piping, supports, and equipment within the Unit 1 MSIV leakage control boundaries meet the appropriate acceptance criteria. The results of this review and evaluation are documented in Enclosure 8 of this submittal. Enclosure 8 provides a description of the ALT pathway including its boundaries, summary of the associated seismic evaluations, and how the criteria of the GE proprietary report NEDC-31858P and the conditions and limitations of the SER were applied.

#### 2.7.1.3 Non-MSIV Leakage Bypass Pathway

For the DBA LOCA, the design basis includes a maximum rate of containment leakage. This primary containment leakage, excluding MSIV leakage, enters the secondary containment (RB) except for 2% that is assumed to bypass the secondary containment. These lines are connected to the condenser, and thus for the evaluation of doses to occupants of the MCR, all of the secondary bypass leakage is assumed to be into the condenser. In a manner similar to the ALT pathway for MSIV leakage, the pathway for secondary containment leakage to the condenser must be able to withstand a design basis earthquake.

Seismic verification for the secondary containment bypass leakage pathway to the condenser has been completed for both units. The following bypass leakage piping is identified as being subject to the seismic verification:

- The portion of the reactor core isolation cooling (RCIC) steam drain line from outside the RB to the main condenser
- The portion of the high-pressure coolant injection (HPCI) steam drain line from outside the RB to the main condenser
- The reactor water cleanup (RWCU) blowdown line to condenser in the TB, from outside the RB to the main condenser

All of the piping is located in the TB, primarily in the condenser bay below the operating floor. All of the identified piping provides a direct flowpath to the main condenser for secondary containment bypass leakage from containment. The piping does not have any branch lines. Therefore, there are no boundary components or isolation valves requiring seismic verification. Also, the piping does not contain any valves that must be positioned in order to provide a flowpath to the main condenser.

The reports for the seismic evaluation of the Units 1 and 2 bypass piping are included as Enclosures 9 and 10, respectively, of this submittal.

**Figure 1. MSIV Alternate Leakage Treatment Pathways**

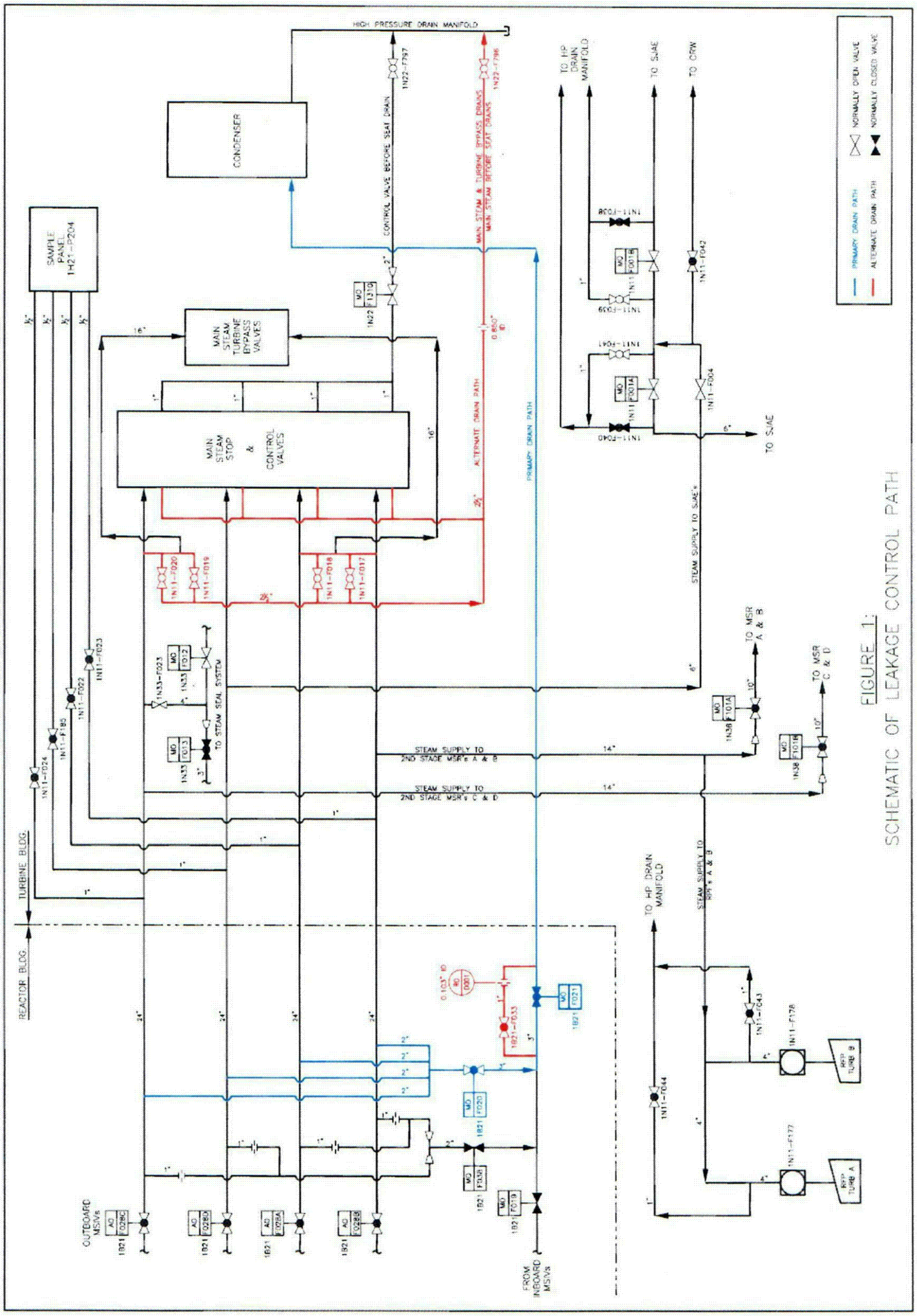


FIGURE 1:
SCHEMATIC OF LEAKAGE CONTROL PATH

### 2.7.2 Standby Liquid Control System

The SLC system is a backup method of manually shutting down the reactor to the cold subcritical mode independent of the control rod system. Availability of SLC is governed by the TSs. The SLC system at HNP is considered a special safety system or safe shutdown system, and not an ESF system. Therefore, the NRC review guideline, "Guidance on the Assessment of a BWR SLC System for pH Control" is used to evaluate the SLC system for its ability to perform its AST function of post-LOCA suppression pool pH control.

Plant procedures will be revised as necessary so that upon detection of high DW radiation associated with the postulated activity release, manual initiation of SLC injection is executed for a LOCA to maintain suppression pool pH at or above 7.0.

SLC is suitably redundant in components and features to assure that its AST function can be accomplished assuming a single active failure. HNP has addressed two potential active failures that could impact the SLC system. The first potential failure is the SLC initiation control switch located on the Unit 1 and Unit 2 panels 1H11-P603 and 2H11-P603, respectively, in the MCR. The second potential failure is one of the two check valves in series on the injection line that are credited to change state to inject the SPB solution.

The SLC initiation control switch is a key-locked, three position switch. The entire assembly is enclosed in a metal cover that provides protection for the contacts. The switch is commonly used in safety and non-safety related applications at HNP and throughout the industry. It is of simple construction with few parts vulnerable to failure. The typical mechanical service life for this switch is estimated to be in the range of 500,000 to 1,000,000 cycles. In the unlikely event of a failure of the control switch to initiate either SLC sub-system, a repair of the switch could be attempted in the MCR, considering that SLC injection is not required for the first 2 hr. An additional compensating action is the ability to install jumpers to overcome failure of the control switch. Procedures will be revised as necessary to address jumper installation for this application.

The injection line check valves are designed to open against full reactor pressure. For the AST function, there would be an even greater differential opening force on the check valves due to the depressurized reactor. Using database searches, no instances of failure to open were found for the injection line check valves. The check valves are considered highly reliable and no compensatory actions are considered necessary to address failure of the component.

Acceptable quality and reliability of the non-redundant active components and the corresponding compensatory actions in the event of failure is demonstrated for the SLC initiation control switch and the injection line check valves.

The environmental conditions for the SLC system have been evaluated with respect to the SLC post-LOCA mission. The SLC system mission time (i.e., the time at which SLC injection is complete) is approximately 6 hr post-LOCA. The post-LOCA RB environmental conditions of interest are temperature and radiation. Pressure and humidity are not environmental factors since the LOCA is in the DW. The post-LOCA RB temperature transient after switching from normal heating, ventilation, and air conditioning (HVAC) to SGTS is slow and the SLC area heat sources are relatively small. It is estimated that 150°F is a reasonable upper bound for the SLC temperature within the first 6 hr of a LOCA. This is considered a mild environment.

Evaluations are performed to determine the dose to SLC components following a LOCA. The total gamma dose, which is the accumulation of normal operating dose over the life of the plant plus 6 hr of dose from the LOCA, is less than 1.0E+04 rad. This is considered to be a mild environment for all components shielded from beta. The only component required for injection that is not shielded from beta is the SLC pump motor. The SLC pump motor has an open drip-proof type enclosure that permits limited exposure of the insulation system to the beta cloud. The total dose (gamma plus beta) to the motor insulation system is determined to be less than 1.0E+04 rad, and is therefore considered to be a mild environment. Cable that is associated with the SLC system is also evaluated and determined to be environmentally qualified for the SLC injection post-LOCA mission.

### 2.7.3 Turbine Building Ventilation

The MCR, as part of the control building, is located in the center of the Units 1 and 2 TBs. The Units 1 and 2 TB ventilation systems are credited in AST with purging the area around the MCR following a LOCA, CRDA, and MSLB. Although the TB ventilation system is not safety-related, a verification of the TB ventilation system is performed using a methodology similar to verifying the MSIV ALT pathway and SLC. From this verification, it can be concluded that the TB ventilation system is highly reliable, and that there exists a high degree of assurance that it will perform its intended function of purging the area around the MCR in the event of one of the abovementioned DBAs.

In addition, a defense-in-depth study on passive ventilation of the TB is conducted to determine expected exhaust flow rates in the absence of forced ventilation. Although passive ventilation is not credited in the analysis, it is shown that without forced TB exhaust, enough ventilation of the TB would exist to maintain MCR doses within regulatory limits, even with unfiltered inleakage into the MCR that is much greater than inleakage actually measured during recent testing.

#### 2.7.3.1 Turbine Building Ventilation System Description

The HNP TB ventilation system is important to operations, and is therefore required to be highly reliable. It is designed to:

- Provide temperature control and air movement control for personnel comfort.
- Optimize equipment performance by the removal of heat dissipated from plant equipment.
- Provide a sufficient quantity of filtered fresh air for personnel.
- Provide for air movement from areas of lesser potential airborne radioactivity to areas of greater potential airborne radioactivity prior to final exhaust.
- Minimize the possibility of exhaust air recirculation into the air intake.
- Minimize the escape of potential airborne radioactivity to the outside atmosphere during normal operation by exhausting air, through a suitable filtration system, from the areas in which a significant potential for radioactive particulates and radioactive iodine contamination exist.

For each unit, air is exhausted from the TB by a duct system to the outside environment via the RB vent plenum by two exhaust fans. The exhaust from the TB is filtered by two 50% capacity filter trains. Each filter train consists of a bank of prefilters, carbon adsorbers, and high efficiency particulate air filters. The carbon adsorber bank is provided with a water deluge system. Only one of the two 100% capacity exhaust fans is normally operating. If the operating exhaust fan fails, the standby exhaust fan starts automatically and an alarm is annunciated in the MCR.

The TB ventilation system incorporates redundancy and other features designed to assure TB operation for normal operation plant conditions. Such features for each unit include:

- A 100% standby supply air fan.
- A 100% standby exhaust air fan.
- Two 50% capacity normally operating charcoal filter trains.
- Provision to adjust supply and exhaust fan flowrates manually so that one filter (50% of normal airflow) can be used during filter maintenance periods.

In the event of a LOCA, CRDA, or MSLB, the appropriate operating procedures will be changed to ensure that TB exhaust ventilation (one of four TB exhaust fans) is initiated within 9 hr of the start of the accident, in accordance with the design basis assumptions for TB ventilation used in the DBA analyses.

The post-accident TB environment is evaluated for the DBAs that credit TB ventilation to ensure that the expected operating environment would not inhibit the functioning of the TB ventilation system. It is concluded that the post-accident environment in each case is a mild environment, with total dose less than 1.0E+04 rad.

2.7.3.2 Failure Analysis

For the TB exhaust system to perform its post-accident function of purging the TB of activity, one of four TB exhaust fans must be able to operate, and the associated exhaust pathway must remain available. The pathway consists of ductwork and a small number of dampers.

Each unit has air-operated inlet isolation valves to the respective TB exhaust filter trains (two parallel filter trains per unit). These are normally open, fail closed valves. Possible active failure modes for these vales are loss of air and loss of power to the normally energized solenoid valve. Either loss of air or power would result in the valves failing closed and stopping flow through the 50% capacity filter trains.

Each unit utilizes air-operated variable pitch inlet dampers. These valves are controlled by normally energized solenoid valves that provide blade pitch control for the TB exhaust fans. Loss of air or power to the solenoid valves would result in closure of the associated inlet damper to allow minimum flow through the in-service TB exhaust fan.

Temperature switches are installed on each unit in the TB exhaust filter train. Upon reaching the high temperature setpoint, contacts close to trip the TB exhaust fans. Loss of power to these switches would result in the contacts staying open, having no effect on the TB exhaust fans.

Loss of power to a single turbine building exhaust fan would result in a low flow annunciation in the MCR and the automatic start of the standby turbine building exhaust fan. Loss of power to both fans in one unit would only result in all turbine building forced exhaust flow being stopped if both exhaust fans in the other unit also failed.

Instruments are used in each TB exhaust system to detect low flow conditions. A low flow condition of the in-service TB ventilation fan would result MCR annunciation. Loss of power to these instruments would result in the loss of ability to automatically start the TB exhaust fans upon a low flow condition. The in-service exhaust fan would continue to run.

Unit 1 utilizes TB exhaust fan outlet dampers. These valves are normally open, fail closed air-operated valves. Loss of air or loss of power to the normally energized solenoid valve would result in valve closure, stopping flow through the Unil 1 TB exhaust filter trains. This failure mechanism is not applicable to Unit 2.

The air for all TB HVAC dampers on both units is supplied by interruptible service instrument air and a combination of three station service air compressors. Loss of instrument air can only occur as the result of one the following:

- A major line break in the compressed air system, or
- The mechanical or electrical failure of the normal instrument air supply, or
- A major dryer failure.

No single failures exist that would impact the TB exhaust capability of both units. The only failure mechanism that could affect both units is a seismic event. The air piping system supplies both safety and non-safety/non-seismic systems. A failure of the air systems of both Unit 1 and Unit 2 would render both TB HVAC systems incapable of performing their required exhaust functions. A modification will be completed to ensure that a loss of air event does not render both TB exhaust systems incapable of operating. This modification will be implemented by December 31, 2009.

No common power supplies exist through the start-up auxiliary transformers 1D and 2D in which failure would result in the loss of the TB ventilation exhaust capability for both units. A discussion of offsite power reliability follows in Section 2.7.3.3.

The motor control center (MCC) panels utilized for TB ventilation are of a robust design. Included in the HNP program resolution to GL 87-02 for NRC Unresolved Safety Issue (USI) A-46, MCC panels 1R25-S037, 1R25-S065, and 2R25-S065 were verified as capable to function after a design basis earthquake using the Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure (GIP), Revision 2, corrected February 14, 1992, as clarified and interpreted by NRC Supplemental Safety Analysis Report No. 2. Using the same methodology, all conduits and cable raceways in the RBs, control buildings, and east cableway were verified as capable to function after a design basis earthquake.

MCC panels 1R25-S120, 1R24-S016, 2R25-S106 and 2R24-S016 were not included in the program resolution for NRC USI A-46, but are similar to the panels evaluated and are expected to perform similarly during a seismic event. The panels will be evaluated using the same SQUG methodology, beginning with walkdowns during the upcoming 2007 and 2008 outages, to verify that they are seismically adequate to withstand the appropriate DBE. The walkdowns will be completed by May 31, 2008.

#### 2.7.3.3 Offsite Power Reliability

HNP has a robust offsite power supply, consisting of four 500-kV transmission lines and four 230-kV transmission lines. A ring bus switching scheme is used for the 500-kV switchyard, and a breaker-and-a-half scheme is utilized for the 230-kV switchyard. Three physically independent 230-kV circuits are provided from the switchyard to startup auxiliary transformers 1C, 1D, 2C, and 2D.

Each transmission line is protected with two protective relaying systems: one primary system and one secondary system. Each power circuit breaker is equipped with two separate trip coils, primary and secondary. These components are connected so that each protective function is redundant, and the loss of any component in one relaying protective scheme, including loss of its battery, in no way affects the proper functioning of the other protective scheme. Each transmission line and both switchyards are equipped with overhead static wires as a designed lightning protection system.

Physical separation, the ring bus, breaker-and-a-half switching schemes, redundant switchyard protection systems, and transmission system design based on load flow and stability studies minimize simultaneous failure of all offsite power sources in compliance with GDC 17.

LOSP data were reviewed from an Idaho National Engineering and Environmental Laboratory (INEEL) document, Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1986-2003 (Reference 6). Data for LOSP events are divided into five categories: Plant Centered, Switchyard Centered, Grid Related, Severe Weather Related, and Extreme Severe Weather Related. The weather related categories are primarily comprised of data from plants exposed to severe ice storms, heavy snow, extreme salt spray, direct hurricane damage, and high winds. Coastal plants, or plants residing near large bodies of water such as the Great Lakes, are much more likely to be affected by high winds than other plants. HNP is not subject to regular input from these severe weather phenomena, although it is subject to tornadoes. This skew on the LOSP data with regard to HNP would tend to make the conclusions in the INEEL report very conservative. Despite the conservatisms, data on LOSP duration that included all five categories of LOSP events were evaluated. The data conclude that for a LOSP event, the probability of the duration exceeding 8 hr is 0.122 (12.2%). Removing the inherent conservatism, the probability would be lower.

The mitigation function for the TB ventilation system is not required until 9 hr after the initiation of the accident (LOCA, CRDA, or MSLB). Assuming a LOSP coincident with the accident, there is ample time to restore offsite power and initiate TB purging via one exhaust fan of the TB ventilation system within the 9 hr.

2.7.3.4 Turbine Building Exhaust Ductwork Seismic Verification

The HNP Units 1 and 2 TB HVAC exhaust ductwork systems were seismically verified to remain in place and maintain exhaust air flow from the TB though the TB exhaust filters to the exhaust stack for the HNP design basis earthquake. The seismic verification methodology is based on earthquake experience data. This methodology is provided in the Electric Power Research Institute (EPRI) Technical Report 1007896 "Seismic Evaluation Guidelines for HVAC Duct and Damper Systems," April 2003. The major steps in this methodology are similar to those provided in the SQUG GIP for raceway systems. These steps are documentation review, in-plant screening walkdowns, analytical review of selected duct runs and supports, and identification and resolution of conditions that do not meet the screening or analysis criteria.

The SQUG methodology was previously used in the HNP resolution to USI A-46 and is an accepted verification methodology. A similar methodology, based on the application of earthquake experience data, was used to verify the seismic adequacy of the HNP TB HVAC system.

SNC had both Units 1 and 2 TB HVAC exhaust ductwork seismic verifications performed by ABS Consulting, the contractor that developed this seismic verification methodology for HVAC systems and authored the EPRI technical report. The seismic verification reports, which consist of the scope of the verifications, methodology, walkdown summary, analytical review, and outlier summary, are included for

Units 1 and 2 as Enclosures 11 and 12, respectively, of this submittal. A summary of the resolution of outliers for both units is presented in the following paragraphs.

Table 4.1 in each of the TB exhaust ductwork seismic verification reports (Enclosures 11 and 12) summarizes the outliers identified as a result of the TB exhaust ductwork walkdowns.

The Unit 1 report (Enclosure 11) identifies four outliers. As detailed in Section 6 of the report, outliers Nos. 1 and 3 were resolved by analysis, outlier No. 2 has been resolved through repair, and outlier No. 4 has been resolved via modification.

The Unit 2 report (Enclosure 12) identifies seven outliers. As detailed in Section 6 of the report, outliers Nos. 2, 3, 5, 6 and 7 were resolved by analysis. Outliers Nos. 1 and 4 require repairs and have been entered into the Corrective Action Program. Resolution of outliers Nos. 1 and 4 is scheduled to be completed by November 28, 2006.

Since this is the first application of the EPRI guidelines and the approach requires the use of engineering judgment, SNC had an independent peer review performed on the EPRI guidelines (Reference 7) and later on the application of these guidelines for the seismic verification of the HNP Unit 1 TB exhaust HVAC system (Reference 8). Both peer reviews were performed by Dr. R. P. Kennedy, an acknowledged industry expert. Dr. Kennedy served as chairman of the five-member independent Senior Seismic Review and Advisory Panel, which provided considerable technical review and input during the development of the SQUG approach for evaluating the seismic adequacy of 20 classes of equipment plus cable and conduit raceway systems and their supports. The panel unanimously endorsed the SQUG approach for use on existing components in existing nuclear power plants. Dr. Kennedy also served on a four-member independent panel established by the NRC to provide advice on the use of the earthquake experience based approach for the seismic qualification of new equipment, cable trays, and HVAC duct systems in new plants.

Recommendations from the peer review of the EPRI guidelines were incorporated in the application of the guidelines for seismic verification of the HNP Unit 1 TB exhaust ductwork. The peer review of the seismic verification of the HNP Unit 1 TB exhaust ductwork concluded that the seismic verification was a very thorough and competent evaluation and fully concurs with the conclusions.

#### 2.7.3.5 Passive Ventilation of the Turbine Building

A defense-in-depth study is performed to determine an estimate of the passive, wind-driven ventilation of the TB. The study is not prepared in conformance with the standards of 10 CFR 50, Appendix B, and as such, the results of the study are not used in any of the design basis analyses or licensing basis. However, analytical techniques are used that are consistent with industry practices for estimating HVAC requirements for commercial and industrial buildings.

The methodology applied for determining the magnitude of passive ventilation of the TB contains four elements:

- Preparation of a cumulative temporal distribution of wind speed for each of the sixteen cardinal, intercardinal, and bisecting wind directions.
- Determination of the pressure coefficient ($C_p$) spatial distribution for each of the sixteen cardinal, intercardinal, and bisecting wind directions.

- Calculation of the wind-generated building internal pressure and cumulative temporal distribution of passive ventilation rate for each of the sixteen cardinal, intercardinal, and bisecting wind directions.
- Combining the cumulative temporal distributions of passive ventilation rate for each of the sixteen cardinal, intercardinal, and bisecting wind directions into a single temporal distribution based on wind direction probability. This final distribution provides the fraction of time that the ventilation rate is less than a given value due to wind-generated pressures around the building. "Stack effect" (i.e., the natural ventilation brought about by temperature differences between the inside and the outside of the building) is conservatively ignored.

HNP FSAR meteorological data are used for the study (Reference 9, Table 2.3-14, Joint Frequency Table of Wind Speed and Direction). The pressure coefficient ($C_p$) spatial distribution is determined using the CpCalc+ code developed by Politecnico di Torino (Turin, Italy) for the European Union. Once the distributed $C_p$ values are known for each wind direction, the ventilation flow is calculated for the partially-open railway doors on the east façade of the TB (at the extreme north and south ends) and for leakage through the pre-cast concrete panel construction of the walls. Flow though the walls is based on the following expression:

$$ELA_4 = Q(4)\sqrt{\frac{\rho}{8}}$$

where Q(4) is the volumetric flow in $m^3$/sec for a 4 Pa pressure difference (approximately one foot of air) across the façade, $ELA_4$ is the equivalent leakage area (in $cm^2/m^2$ of wall area) under the same conditions, and $\rho$ is the density of air in $kg/m^3$. $ELA_4$ values may be found in Persily, A.K. and Ivy, E. M., "Input Data for Multizone Airflow and IAQ Analysis," NISTIR 6585, January 2001, for many different types of construction. The value used for the HNP TB is 4.0 $cm^2/m^2$.

By way of illustrating the nature of the leak path through the turbine building façade, one may note that the pre-cast wall panels used on the turbine building are approximately 7 m x 2 m or 14 $m^2$ with a perimeter of approximately 1800 cm. For a total effective leakage area of approximately 56 $cm^2$ per panel (4 $cm^2/m^2$ x 14 $m^2$), the effective average joint opening would be about 0.03 cm and the actual joint opening (assuming a head loss coefficient of 3) would be approximately 0.05 cm or 0.5 mm. For adjacent panels, the seam opening would be twice that or approximately 1 mm. Such small seam openings at the construction joints would be barely discernable. Flow through the wall is assumed to be proportional to $\Delta h^{0.65}$ where $\Delta h$ is the head loss of air across the wall. This proportionality is typically assumed for this type of analysis.

The railway roll-up doors are 20 ft wide and are normally kept open to a height of 5 ft. They are blocked by grating that is assumed to be 80% free area. Assuming a head loss coefficient of 3, the effective area of each partially-open railway door is approximately 46 $ft^2$. Flow through the partially-open railway roll-up doors is assumed to be proportional to the square-root of the head loss, as for an orifice.

Using this model, the following temporal distribution of ventilation rates in the absence of forced ventilation for the TB has been calculated:

**Table 36. Temporal Distribution of Turbine Building Ventilation Rates**

| Percentile | Volumetric Flow Rate |
|---|---|
| $5^{th}$ | 3,300 cfm |
| $10^{th}$ | 4,900 cfm |
| $20^{th}$ | 7,000 cfm |
| $40^{th}$ | 10,700 cfm |

For study purposes, following the logic of $\chi/Q$ development described in NRC regulatory guidance, it would be reasonable to apply the $5^{th}$ percentile value of 3,300 cfm for the first 8 hr of the MCR dose analysis, 4,900 cfm for the next 16 hr of MCR dose analysis, 7,000 cfm for the next 72 hr of MCR dose analysis, and 10,700 cfm for the remainder of the 30 day duration of the MCR dose analysis. The radiological dose consequences for the MCR for the LOCA, CRDA, and MSLB, assuming only passive ventilation of the TB, are compared to the doses calculated in the design basis analyses using TB exhaust flow of 15,000 cfm (provided by one TB exhaust fan) in Table 37. For the passive ventilation cases, the maximum MCR unfiltered inleakage that could be tolerated and remain within regulatory dose limits is used, instead of the design basis value of 115 cfm.

The results of the defense-in-depth study conclude that significant air exhaust of the TB from natural, wind-driven ventilation would maintain the radiological dose consequences to occupants of the MCR within regulatory limits, with no forced TB exhaust for the full 30-day accident duration for the three DBAs (LOCA, CRDA, and MSLB) that credit TB ventilation. These assume MCR unfiltered inleakage results that are less conservative than the DBA analyses, but with significant margin to measured inleakage results from recent tracer gas inleakage testing.

**Table 37. MCR Dose with Passive Ventilation**

| | Passive Ventilation | | Design Basis | Limit (rem TEDE) |
|---|---|---|---|---|
| | Maximum Inleakage | MCR Dose (rem TEDE) | MCR Dose (rem TEDE) | |
| LOCA | 70 cfm | 4.9 | 4.9 | 5 |
| CRDA | 85 cfm | 4.9 | 3.8 | 5 |
| MSLB | 130 cfm | 4.9 | 3.9 | 5 |

Note: Design basis assumes 115 cfm unfiltered inleakage.

## 2.8 NUREG-0737 Evaluation

The inputs and assumptions utilized in the NUREG-0737 evaluation include the AST plant-specific fission product inventories and other applicable inputs as described in Section 2.5.

### 2.8.1 Post-Accident Access Shielding

Plant calculations used in support of plant post-accident vital area[5] access, prepared in accordance with NUREG-0737, Item II.B.2, are evaluated for impact by AST. The implementation of AST results in new activities to be performed post-accident in vital areas. The new activities, applicable post-LOCA, are isolation of the MSLs and establishment of a pathway for MSIV leakage to the main condenser. This requires access to several locations in the TB. A dose evaluation is performed using conservative estimates for walking speed and valve operation times.

The evaluation considered the comparative radiation levels from AST and the existing TID-14844 methodology source term (which is based on reactor power of 2,537 MWt). The results of the evaluation conclude that the current source term used for shielding remains bounding, even with the increase in power to 2,818 MWt and two-year fuel cycles. The NUREG-0737 Item II.B.2 review of plant shielding and environmental qualification of equipment previously completed remains applicable for AST. The dose evaluation of the new activities required to isolate the MSLs and establish the MSIV leakage pathway to the main condenser indicates that these activities can be completed with operator exposures of 5 rem TEDE or less. The evaluated doses continue to meet GDC 19 criteria and NUREG-0737 Item II.B.2.

### 2.8.2 Post-Accident Radiation Monitor

The containment high-range radiation monitors used to monitor post-accident primary containment radiation levels are evaluated for the impact of AST. The monitors continue to provide their design function and envelop the projected radiation exposure rates. Accident radiation monitoring instrumentation continues to meet the requirements of NUREG-0737 Item II.F.1.

In addition, the control room intake radiation monitor setpoint is evaluated for AST. The evaluation has determined that the current setpoint will alarm and initiate the MCREC system at the start of any of the four DBAs analyzed for AST.

### 2.8.3 Leakage Control

The DBA LOCA control room and TSC dose analysis, as well as that for offsite doses, explicitly considers the effects of coolant leakage outside the primary containment (ESF leakage), satisfying the requirements of NUREG-0737, Item III.D.1.1.

### 2.8.4 Control Room and TSC Radiation Protection

The radiological dose impacts to the MCR and TSC have been specifically calculated for each of the four DBAs analyzed for AST implementation (NUREG-0737, Item III.A.1.2 and III.D.3.4). In addition, shine from contained sources is also evaluated. The current MCR dose from secondary containment shine is a bounding value for AST. The results of these analyses are presented in Sections 2.5.2.3 (LOCA), 2.5.3.3 (FHA), 2.5.4.3 (CRDA), and 2.5.5.3 (MSLB). The evaluated doses remain less than 5 rem TEDE.

[5] As defined by Reference 10, a vital area is any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident.

# 3. CONCLUSIONS

HNP is proposing a full-scope implementation of the AST. Application of the AST methodology for the four DBAs identified in the HNP FSAR that could result in significant control room and offsite doses has been completed using analysis methods and assumptions consistent with the conservative guidance of RG 1.183. This analysis has demonstrated that doses to occupants of the MCR and the TSC, and offsite (EAB and LPZ) doses remain within regulatory limits.

# 4. REFERENCES

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4. E. C. Beahm, C. F. Weber, T. S. Kress, and G. W. Parker, "Iodine Chemical Forms in LWR Severe Accidents," NUREG/CR-5732, ORNL/TM-11861, April 1992.

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16. Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.

17. Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993.

18. Generic Implementation Procedure (GIP) For Seismic Verification of Nuclear Plant Equipment, Revision 3A, Seismic Qualification Utility Group, December 2001.

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20. Hatch Unit 1 Technical Specifications, Amendment 248.

21. Hatch Unit 2 Technical Specifications, Amendment 192.

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24. NRC Generic Letter 2003-01, "Control Room Habitability," June 12, 2003.

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

26. NRC Regulatory Guide 1.183, "Alternative Source Terms for Evaluating Design Basis Accidents At Nuclear Power Reactors," July 2000.

27. NRC Regulatory Issues Summary 2006-04, "Experience with Implementation of Alternative Source Terms," March 7, 2006.

28. NRC Standard Review Plan Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms," Revision 0, July 2000.

29. ORIGEN2 Computer Code, Oak Ridge National Laboratory.

30. Safety Evaluation on the Resolution of Unresolved Safety Issue A-46 at Edwin I. Hatch Plant, Units 1 and 2 (TAC Nos. N69451 and M69452), enclosure to letter from Leonard N. Olshan, Project Manager, NRC to Mr. H. L. Summer, Jr., Southern Nuclear Operating Company, September 24, 1998.

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# APPENDIX A: Regulatory Guide 1.183 Conformance Matrix

| Conformance with Regulatory Guide 1.183 - Main Sections | | | |
|---|---|---|---|
| RG Sec | RG Position | HNP Position | Comments |
| 3.1 | The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 (Ref. 17) or ORIGEN-ARP (Ref. 18). Core inventory factors (Ci/MWt) provided in TID14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels. | Conforms. | Core power accounts for 0.5% uncertainty. 10% margin is added to core fission product inventory to allow for future fuel changes or power uprates. |
| 3.1 | For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. | Conforms. | |
| 3.2 | The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are listed in Table 1 for BWRs and Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1. | Conforms. | The release fractions from Table 1 are used. |
| 3.2 | For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor.<br><br>Table 3<br>Non-LOCA Fraction of Fission Product Inventory in Gap<br>**Group** — **Fraction**<br>I-131 — 0.08<br>Kr-85 — 0.10 | Conforms. | Release fraction of 0.1 is used for I-131, Other Noble Gases, and Other Halogens in the CRDA analysis. |

| Conformance with Regulatory Guide 1.183 - Main Sections | | | |
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| RG Sec | RG Position | HNP Position | Comments |
| | Other Noble Gases 0.05<br>Other Halogens 0.05<br>Alkali Metals 0.12 | | |
| 3.3 | Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase. For non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage. | Conforms. | The BWR durations from Table 4 are used for the LOCA. |
| 3.4 | Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses. | Conforms. | |
| 3.5 | Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details. | Conforms. | |
| 3.6 | The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy | Conforms. | |

| Conformance with Regulatory Guide 1.183 - Main Sections | | | |
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| RG Sec | RG Position | HNP Position | Comments |
| | deposition, for estimating fuel damage for the purpose of establishing radioactivity releases. | | |
| 4.1 | Offsite Dose Consequences | Conforms. | |
| 4.1.1 | The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides, that are significant with regard to dose consequences and the released radioactivity. | Conforms. | |
| 4.1.2 | The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE. | Conforms. | Dose Conversion Factors from Federal Guidance Report 11 are used. |
| 4.1.3 | For the first 8 hours, the breathing rate of persons offsite should be assumed to be $3.5 \times 10^{-4}$ cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be $1.8 \times 10^{-4}$ cubic meters per second. After that and until the end of the accident, the rate should be assumed to be $2.3 \times 10^{-4}$ cubic meters per second. | Conforms. | |
| 4.1.4 | The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 21), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE. | Conforms. | Dose Conversion Factors from Federal Guidance Report 12 are used. |
| 4.1.5 | The TEDE should be determined for the most limiting person at the EAB. The maximum | Conforms. | |

| Conformance with Regulatory Guide 1.183 - Main Sections | | | |
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| RG Sec | RG Position | HNP Position | Comments |
| | EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67. The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see also Table 6). | | |
| 4.1.6 | TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67. | Conforms | |
| 4.1.7 | No correction should be made for depletion of the effluent plume by deposition on the ground. | Conforms. | |
| 4.2.1 | The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include:<br>• Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility,<br>• Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope,<br>• Radiation shine from the external radioactive plume released from the facility,<br>• Radiation shine from radioactive material in the reactor containment,<br>• Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters. | Conforms. | |
| 4.2.2 | The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these | Conforms. | Main condenser release (from MSIV leakage and secondary containment |

| Conformance with Regulatory Guide 1.183 - Main Sections | | | |
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| RG Sec | RG Position | HNP Position | Comments |
| | assumptions would result in non-conservative results for the control room. | | bypass leakage) is to turbine building, to maximize control room dose since the control room is within the turbine building. |
| 4.2.3 | The models used to transport radioactive material into and through the control room, and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel. | Conforms. | |
| 4.2.4 | Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. Such features may include control room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP (Ref. 3) and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post-accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 25), for guidance. | Conforms. | Pressurization and intake filtration are credited. |
| 4.2.5 | Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis. | Conforms. | |
| 4.2.6 | The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days. For the duration of the event, the breathing rate of this individual should be assumed to be $3.5 \times 10^{-4}$ cubic meters per second. | Conforms. | |
| 4.2.7 | Control room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, $DDE_\infty$, to a | Conforms. | A rigorous analysis treating the control room as a 148' x 66' x 16' volume was performed. Based on this rigorous |

| Conformance with Regulatory Guide 1.183 - Main Sections | | | |
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| RG Sec | RG Position | HNP Position | Comments |
| | finite cloud dose, $DDE_{finite}$, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room (Ref. 22). $DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173}$ | | analysis, an additional 0.5 correction factor is applied to the correction factor calculated from the expression to the left. |
| 4.3 | The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 2). Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide. | Conforms. | NUREG-0737 analysis using AST was completed. |
| 4.4 | The radiological criteria for the EAB, the outer boundary of the LPZ, and for the control room are in 10 CFR 50.67. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. The control room criterion applies to all accidents. For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6.<br>The acceptance criteria for the various NUREG-0737 (Ref. 2) items generally reference General Design Criteria 19 (GDC 19) from Appendix A to 10 CFR Part 50 or specify criteria derived from GDC-19. These criteria are generally specified in terms of whole body dose, or its equivalent to any body organ. For facilities applying for, or having received, approval for the use of an AST, the applicable criteria should be updated for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii). | Conforms. | |
| 5.1.1 | The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. | Conforms. | |

| Conformance with Regulatory Guide 1.183 - Main Sections | | | |
|---|---|---|---|
| RG Sec | RG Position | HNP Position | Comments |
| 5.1.2 | Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences. | Does not conform. | Credit is taken for the alternate leakage treatment pathway via the main condenser, the standby liquid control system (a safe shutdown system, not an ESF system), and turbine building ventilation. See Section 2.7 for the justification of this credit. |
| 5.1.3 | The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis. | Conforms. | |
| 5.1.4 | Licensees should ensure that analysis assumptions and methods are compatible with the AST and the TEDE criteria. | Conforms. | |
| 5.2 | Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST. | Conforms. | |
| 5.3 | Atmospheric dispersion values (X/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. Methodologies that have been used for determining X/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19" (Refs. 6, 7, 22, and 28).<br>The methodology of the NRC computer code ARCON96 (Ref 26) is generally acceptable to the NRC staff for use in determining control room X/Q values. | Conforms. | Only control room atmospheric dispersion factors were re-calculated for AST. ARCON96 was used for updating these values. These values are also applied to the TSC, since the control room values are bounding. |

| Conformance with Regulatory Guide 1.183 – Appendix A (LOCA) | | | |
|---|---|---|---|
| App Sec | RG Position | HNP Position | Comments |
| 1 | Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide. | Conforms. | See main Sections 3.1 to 3.4 for more information. |
| 2 | If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species, including those from iodine re-evolution, for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids and bases created during the LOCA event, e.g., radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. | Conforms. | |
| 3.1 | The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs or the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell. The release into the containment or drywell should be assumed to terminate at the end of the early in-vessel phase. | Conforms. | Flow from the DW to the torus prior to the assumed core quench at 2 hours is conservatively ignored. Post-reflood flow from DW to torus is considered for a period of time to bring about a near uniform distribution of activity. This modeling is conservative relative to the assumption of a well-mixed containment post-reflood. |
| 3.2 | Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are | Conforms. | STARNAUA is used instead of NUREG/CR- |

| Conformance with Regulatory Guide 1.183 – Appendix A (LOCA) | | | |
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| App Sec | RG Position | HNP Position | Comments |
| | described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 (Ref. A-1) and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Ref. A-2). The latter model is incorporated into the analysis code RADTRAD (Ref. A-3). | | 6189 to predict natural deposition of aerosol prior to the start of sprays. Also refer to item 3.3 below. |
| 3.3 | Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP (Ref. A-1) may be credited. Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays"1 (Ref. A-4). This simplified model is incorporated into the analysis code RADTRAD (Refs. A-1 to A-3). | Conforms. | STARNAUA is used instead of the methods cited. STARNAUA combines the effects of both natural deposition and sprays. |
| 3.3 | The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown. | Conforms. | DW is assumed to be well-mixed based on the fact that the DW is sufficiently small and the spray flowrate is sufficiently large (i.e., the ratio of spray flow to volume sprayed is 20-40 times larger for the HNP DW than for a typical sprayed region of a PWR) that mixing by momentum exchange alone (between the droplets and the atmosphere) will keep the DW well-mixed; i.e., natural convection will play no noticeable role. |

| Conformance with Regulatory Guide 1.183 – Appendix A (LOCA) | | | |
|---|---|---|---|
| App Sec | RG Position | HNP Position | Comments |
| 3.3 | The SRP sets forth a maximum decontamination factor (DF) for elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays. The maximum activity to be used in determining the DF is defined as the iodine activity in the columns labeled "Total" in Tables 1 and 2 of this guide multiplied by 0.05 for elemental iodine and by 0.95 for particulate iodine (i.e., aerosol treated as particulate in SRP methodology). | Conforms. | Maximum iodine DF is based on projected pH of suppression pool, not on the SRP. STARNAUA does not explicitly reduce the removal rate for particulate by a factor of 10 when a DF of 50 is reached, but the code does take into account the small size of the remaining particles, and the same removal rate reduction effect is realized. |
| 3.4 | Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. A-5 and A-6). The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed. | Not applicable. | |
| 3.5 | Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool (Ref. 7). Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7. | Conforms. | Suppression pool scrubbing not credited. |
| 3.6 | Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above, should be evaluated on an individual case basis. See Section 6.5.4 of the SRP (Ref. A-1). | Not applicable. | |

| Conformance with Regulatory Guide 1.183 – Appendix A (LOCA) | | | |
|---|---|---|---|
| App Sec | RG Position | HNP Position | Comments |
| 3.7 | The primary containment (i.e., drywell for Mark I and II containment designs) should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the technical specification leak rate. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50% of the technical specification leak rate. Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition as defined by technical specifications. | Conforms. | |
| 3.8 | If the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100% of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the technical specification reactor coolant system equilibrium activity. Iodine spikes need not be considered. If the purge system is not isolated before the onset of the gap release phase, the release fractions associated with the gap release and early in-vessel phases should be considered as applicable. | Not applicable. | |
| 4.1 | Leakage from the primary containment should be considered to be collected, processed by engineered safety feature (ESF) filters, if any, and released to the environment via the secondary containment exhaust system during periods in which the secondary containment has a negative pressure as defined in technical specifications. Credit for an elevated release should be assumed only if the point of physical release is more than two and one-half times the height of any adjacent structure. | Conforms. | |
| 4.2 | Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in technical specifications. | Conforms. | |
| 4.3 | The effect of high wind speeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5% of the total number of | Conforms. | The 95$^{th}$ percentile wind speed for HNP is approximately 12 mph. |

| Conformance with Regulatory Guide 1.183 – Appendix A (LOCA) | | | |
|---|---|---|---|
| App Sec | RG Position | HNP Position | Comments |
| | hours in the data set. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded only 5% or 95% of the total numbers of hours in the data set, whichever is conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5%). | | The dynamic pressure for such a wind speed is 0.07 in wg. Even if the minimum wind pressure coefficient on the reactor building façade were as negative as -1.0, the minimum pressure on the surface of the reactor building (-0.07 in wg relative to ambient static pressure) would still be greater than the Technical Specification surveillance limit of at least -0.2 in wg (with approximately 200% margin). |
| 4.4 | Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Credit for mixing, if found to be appropriate, should generally be limited to 50%. This evaluation should consider the magnitude of the containment leakage in relation to contiguous building volume or exhaust rate, the location of exhaust plenums relative to projected release locations, the recirculation ventilation systems, and internal walls and floors that impede stream flow between the release and the exhaust. | Conforms. | 50% of the reactor building volume is credited for dilution. |
| 4.5 | Primary containment leakage that bypasses the secondary containment should be evaluated at the bypass leak rate incorporated in the technical specifications. If the bypass leakage is through water, e.g., via a filled piping run that is maintained full, credit for retention of iodine and aerosols may be considered on a case-by-case basis. Similarly, deposition of aerosol radioactivity in gas-filled lines may be considered on a | Conforms. | |

| Conformance with Regulatory Guide 1.183 – Appendix A (LOCA) | | | |
|---|---|---|---|
| App Sec | RG Position | HNP Position | Comments |
| | case-by-case basis. | | |
| 4.6 | Reduction in the amount of radioactive material released from the secondary containment because of ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6). | Conforms. | |
| 5.1 | With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Tables 1 and 2 of this guide) should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are nonconservative with regard to the buildup of sump activity. | Conforms. | |
| 5.2 | The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737 (Ref. A-8), would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank. | Conforms. | As there is no technical specification limit, a conservatively high leakage rate of 10 gpm is assumed. ESF leakage is assumed to begin at the time DW sprays are started. |
| 5.3 | With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase. | Conforms. | |
| 5.4 | If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. | Not applicable. | |

| Conformance with Regulatory Guide 1.183 – Appendix A (LOCA) | | | |
|---|---|---|---|
| App Sec | RG Position | HNP Position | Comments |
| 5.5 | If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates. | Conforms. | A release fraction of 10% is assumed. |
| 5.6 | The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6). | Conforms. | Credit is taken for holdup and dilution of ESF leakage in reactor building and for release through SGTS filters via plant stack. |
| 6.1 | For the purpose of this analysis, the activity available for release via MSIV leakage should be assumed to be that activity determined to be in the drywell for evaluating containment leakage (see Regulatory Position 3). No credit should be assumed for activity reduction by the steam separators or by iodine partitioning in the reactor vessel. | Conforms. | |
| 6.2 | All the MSIVs should be assumed to leak at the maximum leak rate above which the technical specifications would require declaring the MSIVs inoperable. The leakage should be assumed to continue for the duration of the accident. Postulated leakage may be reduced after the first 24 hours, if supported by site-specific analyses, to a value not less than 50% of the maximum leak rate. | Conforms. | The full technical specification maximum combined leakage for all MSIVs is conservatively assumed through the failed line. |
| 6.3 | Reduction of the amount of released radioactivity by deposition and plateout on steam system piping upstream of the outboard MSIVs may be credited, but the amount of reduction in concentration allowed will be evaluated on an individual case basis. Generally, the model should be based on the assumption of well-mixed volumes, but other models such as slug flow may be used if justified. | Conforms. | Impaction credited as well as sedimentation (calculated with STARNAUA). Well-mixed volumes are assumed. |
| 6.4 | In the absence of collection and treatment of releases by ESFs such as the MSIV leakage control system, or as described in paragraph 6.5 below, the MSIV leakage should be | Conforms. | Since the control room is located within the turbine |

| Conformance with Regulatory Guide 1.183 – Appendix A (LOCA) | | | |
|---|---|---|---|
| App Sec | RG Position | HNP Position | Comments |
| | assumed to be released to the environment as an unprocessed, ground- level release. Holdup and dilution in the turbine building should not be assumed. | | building, it is more conservative to assume holdup in the turbine building than to assume direct release to the environment when calculating control room dose. |
| 6.5 | A reduction in MSIV releases that is due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, may be credited if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake (SSE). The amount of reduction allowed will be evaluated on an individual case basis. References A-9 and A-10 provide guidance on acceptable models. | Conforms. | Particulate and elemental iodine deposition is credited in the piping and in the main condenser. Particulate deposition is calculated using STARNAUA. |
| 7 | The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Reduction in the amount of radioactive material released via ESF filter systems may be taken into account provided that these systems meet the guidance in Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6). | Not applicable. | |

| Conformance with Regulatory Guide 1.183 – Appendix B (FHA) | | | |
|---|---|---|---|
| App Sec | RG Position | HNP Position | Comments |

| Conformance with Regulatory Guide 1.183 – Appendix B (FHA) | | | |
|---|---|---|---|
| App Sec | RG Position | HNP Position | Comments |
| 1 | Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide. | Conforms. | |
| 1.1 | The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered. | Conforms. | |
| 1.2 | The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums. | Conforms. | |
| 1.3 | The chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water. Because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This is assumed to occur instantaneously. The NRC staff will consider, on a case-by-case basis, justifiable mechanistic treatment of the iodine release from the pool. | Conforms. | |
| 2 | If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method (Ref. B-1). | Conforms. | Water depth is 21 ft. A smaller decontamination factor is calculated using the guidance of the regulatory guide. |
| 3 | The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained | Conforms. | |

| Conformance with Regulatory Guide 1.183 – Appendix B (FHA) | | | |
|---|---|---|---|
| App Sec | RG Position | HNP Position | Comments |
| | by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor). | | |
| 4 | Fuel handling accidents within the fuel building. | Not applicable. | |
| 5.1 | If the containment is isolated during fuel handling operations, no radiological consequences need to be analyzed. | Conforms. | Containment is not isolated. Radiological consequences analyzed. |
| 5.2 | If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment, no radiological consequences need to be analyzed. | Not applicable. | |
| 5.3 | If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open), the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period. | Conforms. | |
| 5.4 | A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2 and B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses. | Conforms. | ESF filter systems not credited. |
| 5.5 | Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50% of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the reactor cavity and the exhaust plenums. | Conforms. | No credit taken for dilution or mixing in the reactor building. |

| Conformance with Regulatory Guide 1.183 – Appendix C (CRDA) | | | |
|---|---|---|---|
| App Sec | RG Position | HNP Position | Comments |
| 1 | Assumptions acceptable to the NRC staff regarding core inventory are provided in Regulatory Position 3 of this guide. For the rod drop accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and on the assumption that 100% of the noble gases and 50% of the iodines contained in that fraction are released to the reactor coolant. | Conforms. | |
| 2 | If no or minimal fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity (typically 4 µCi/gm DE I-131) allowed by the technical specifications. | Conforms. | Substantial fuel damage is postulated. Coolant activity neglected. |
| 3.1 | The activity released from the fuel from either the gap or from fuel pellets is assumed to be instantaneously mixed in the reactor coolant within the pressure vessel. | Conforms. | |
| 3.2 | Credit should not be assumed for partitioning in the pressure vessel or for removal by the steam separators. | Conforms. | |
| 3.3 | Of the activity released from the reactor coolant within the pressure vessel, 100% of the noble gases, 10% of the iodine, and 1% of the remaining radionuclides are assumed to reach the turbine and condensers. | Conforms. | |
| 3.4 | Of the activity that reaches the turbine and condenser, 100% of the noble gases, 10% of the iodine, and 1% of the particulate radionuclides are available for release to the environment. The turbine and condensers leak to the atmosphere as a ground- level release at a rate of 1% per day for a period of 24 hours, at which time the leakage is assumed to terminate. No credit should be assumed for dilution or holdup within the turbine building. Radioactive decay during holdup in the turbine and condenser may be assumed. | Conforms. | |
| 3.5 | In lieu of the transport assumptions provided in paragraphs 3.2 through 3.4 above, a more mechanistic analysis may be used on a case-by-case basis. Such analyses account | Not applicable. | |

| Conformance with Regulatory Guide 1.183 – Appendix C (CRDA) | | | |
|---|---|---|---|
| App Sec | RG Position | HNP Position | Comments |
| | for the quantity of contaminated steam carried from the pressure vessel to the turbine and condensers based on a review of the minimum transport time from the pressure vessel to the first main steam isolation (MSIV) and considers MSIV closure time. | | |
| 3.6 | The iodine species released from the reactor coolant within the pressure vessel should be assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic. The release from the turbine and condenser should be assumed to be 97% elemental and 3% organic. | Conforms. | |

| Conformance with Regulatory Guide 1.183 – Appendix D (MSLB) | | | |
|---|---|---|---|
| App Sec | RG Position | HNP Position | Comments |
| 1 | Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. | Conforms. | No fuel damage. Release estimate based on coolant activity. |
| 2 | If no or minimal fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity allowed by technical specification. The iodine concentration in the primary coolant is assumed to correspond to the following two cases in the nuclear steam supply system vendor's standard technical specifications. | Conforms. | |
| 2.1 | The concentration that is the maximum value (typically 4.0 μCi/gm DE I-131) permitted and corresponds to the conditions of an assumed pre-accident spike, and the concentration that is the maximum equilibrium value (typically 0.2 μCi/gm DE I-131) permitted for continued full power operation. | Conforms. | 2.0 μCi/g DE I-131 for pre-accident spike; corresponds to maximum technical specification limit. |
| 3 | The activity released from the fuel should be assumed to mix instantaneously and homogeneously in the reactor coolant. Noble gases should be assumed to enter the steam phase instantaneously. | Conforms. | |
| 4.1 | The main steam line isolation valves (MSIV) should be assumed to close in the | Conforms. | |

| Conformance with Regulatory Guide 1.183 – Appendix D (MSLB) | | | |
|---|---|---|---|
| App Sec | RG Position | HNP Position | Comments |
| | maximum time allowed by technical specifications. | | |
| 4.2 | The total mass of coolant released should be assumed to be that amount in the steam line and connecting lines at the time of the break plus the amount that passes through the valves prior to closure. | Conforms. | |
| 4.3 | All the radioactivity in the released coolant should be assumed to be released to the atmosphere instantaneously as a ground-level release. No credit should be assumed for plateout, holdup, or dilution within facility buildings. | Conforms for offsite doses. | For doses to control room, which is located within the turbine building, it is conservatively assumed that activity is released directly into the turbine building, thereby providing a direct inleakage pathway to the control room. |
| 4.4 | The iodine species released from the main steam line should be assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic. | Conforms. | |

# APPENDIX B: Design Inputs and Assumptions for DBA Analyses

| Parameter | Current Value | AST Design Basis Value |
|---|---|---|
| Design Inputs and Assumptions Common to Multiple DBA Analyses | | |
| **Source Terms** | | |
| Core Fission Product Inventory | GE generic fission product inventory | HNP 24-month bounding fission product inventory calculated using ORIGEN2 and multiplied by 1.1 |
| **Volumes and Dimensions** | | |
| Minimum Torus Air Volume | 1.13E5 $ft^3$ for Unit 2 | 1.10E5 $ft^3$ for Unit 2 |
| RB Volume | Not used | 1.38E6 $ft^3$ (Unit 1)<br>1.30E6 $ft^3$ (Unit 2) |
| TB Free Volume | Not used | 6.5E6 $ft^3$ |
| Condenser Volume | 8.32E4 $ft^3$ | 1.72E5 $ft^3$ combined volume of low-pressure turbine and condenser |
| Control Room External Dimensions | Not used | 148 ft x 66 ft x 16 ft |
| Control Room External Shielding | Not used | 2 ft thick concrete walls, 2.5 ft thick concrete roof |
| **MCR / TSC Ventilation** | | |
| MCR Filtered Intake Rate | 400 cfm | 250 cfm |
| Limiting MCR Unfiltered Inleakage (LOCA is Limiting DBA) | 110 cfm | 115 cfm (for LOCA, CRDA, and MSLB) |

| Parameter | Current Value | AST Design Basis Value |
|---|---|---|
| TSC Filter Efficiency | 99% for all isotopes except noble gases for intake and recirculation | 90% for all isotopes except noble gases for intake and recirculation |
| TSC Unfiltered Inleakage | 0 cfm | 10,000 cfm |
| **TB Ventilation** | | |
| TB Fan Start Time | Not used | 9 hr after accident initiation |
| TB Exhaust Rate | Not used | 15,000 cfm |
| **Atmospheric Dispersion** | | |
| MCR and TSC $\chi$/Q for RB Vent Release at Ground Level | Unit 1:<br>9.90E-4 sec/m$^3$ (0 – 2 hr)<br>3.97E-4 (2 – 8 hr)<br>4.30E-4 (8 – 24 hr)<br>3.22E-4 (24 – 96 hr)<br>2.62E-4 (96 – 720 hr)<br>Unit 2:<br>1.26E-3 sec/m$^3$ (0 – 2 hr)<br>3.87E-4 (2 – 8 hr)<br>4.17E-4 (8 – 24 hr)<br>3.56E-4 (24 – 96 hr)<br>2.37E-4 (96 – 720 hr) | Both Units:<br>1.41E-3 sec/m$^3$ (0 – 2 hr)<br>1.08E-3 (2 – 8 hr)<br>4.70E-4 (8 – 24 hr)<br>3.54E-4 (24 – 96 hr)<br>2.67E-4 (96 – 720 hr) |
| MCR and TSC $\chi$/Q for Release Through Plant Stack | Both units:<br>4.85E-6 sec/m$^3$ (0 – 2 hr)<br>1.17E-6 (2 – 8 hr)<br>9.69E-7 (8 – 24 hr)<br>8.27E-7 (24 – 96 hr)<br>5.49E-7 (96 – 720 hr) | Both units:<br>3.76E-6 sec/m$^3$ (0 – 2 hr)<br>2.88E-6 (2 – 8 hr)<br>7.50E-7 (8 – 24 hr)<br>7.67E-7 (24 – 96 hr)<br>5.04E-7 (96 – 720 hr) |
| LOCA Inputs | | |

| Parameter | Current Value | AST Design Basis Value |
|---|---|---|
| SGTS Flow Rate | Not used | 8,000 cfm (assumes 2 fans aligned to single RB volume) |
| Primary Containment Leakage | 1.2% per day | 1.2% per day (0-24 hr)<br>• Reduce by 40% (24-72 hr)<br>• Reduce by 50% ( > 72 hr) |
| Secondary Containment Bypass Leakage | 0.9% of primary containment leakage | 2.0% of primary containment leakage |
| ESF Leakage | 0 gpm (not modeled) | 10 gpm |
| MSIV Leakage | 250 scfh | 100 scfh total (all modeled from failed line)<br>• Reduce by 40% (24-72 hr)<br>• Reduce by 50% ( > 72 hr) |
| Condenser Leakage | 6.8% per day | Mass balance based on flow into condenser |
| DW Spray Start Time | Not used | 15 min after accident initiation |
| Holdup in TB | Not used | Yes (for MCR doses) |
| Use of KI Tablets | Taken by operators to lower thyroid dose | Not used |
| FHA Inputs | | |
| Number of Fuel Rods per Bundle | 62 | 87.3 |
| Number of Fuel Rods with Cladding Failure | 125 | 172 |
| Minimum Depth of Water Above Damaged Fuel | 23 ft | 21 ft |
| Holdup and Filtration in Secondary Containment | Yes | Both cases evaluated; holdup and filtration in secondary containment is not required |

| Parameter | Current Value | AST Design Basis Value |
|---|---|---|
| EAB and LPZ $\chi/Q$ for Leakage at Ground Level (0 – 120 sec) | 4.1E-4 sec/m$^3$ | 3.1E-4 sec/m$^3$ |
| CRDA Inputs | | |
| Number of Fuel Rods per Bundle | 62 | 87.3 |
| Number of Fuel Rods with Cladding Failure | 763 | 1189 |
| Number of Fuel Rods with Melting | 7 | 11 |
| Radionuclide Percentage Released from Vessel | 100% noble gases<br>10% iodines | 100% noble gases<br>10% iodines<br>1% particulates |
| Radionuclide Percentage Released from Turbine / Condenser | 100% noble gases<br>10% iodines | 100% noble gases<br>10% iodines<br>1% particulates |
| Turbine / Condenser Leak Rate | 0.5% per day | 1% per day |
| Holdup in TB | Not used | Yes (for MCR doses) |
| MSLB Inputs | | |
| Dose Conversion Factors Used to Calculate DE I-131 | RG 1.109 inhalation thyroid | FGR 11 inhalation effective |
| Primary Coolant Iodine Activity | 4.0 μCi/g DE I-131 (Pre-accident I spike)<br>0.2 μCi/g DE I-131 (Equilibrium I activity) | 2.0 μCi/g DE I-131 (Pre-accident I spike)<br>0.2 μCi/g DE I-131 (Equilibrium I activity) |
| Holdup in TB | Not used | Yes (for MCR doses) |
| EAB and LPZ $\chi/Q$ for Leakage at Ground Level (0 – 1 hr) | 4.1E-4 sec/m$^3$ | 3.1E-4 sec/m$^3$ |

Enclosure 2

Edwin I. Hatch Nuclear Plant
Request to Implement an Alternative Source Term

Description and Justification for Proposed Change

## Description and Justification for Proposed Change

### Proposed Change

Southern Nuclear Operating Company (SNC) requests a revision of the Edwin I. Hatch Nuclear Plant (HNP) Operating License by revising the Technical Specifications (TSs) and incorporating an alternative source term (AST) methodology into the facility's licensing basis. The proposed license amendment involves a full scope implementation of an AST methodology by revising the current accident source term and replacing it with an accident source term as prescribed in 10 CFR 50.67.

AST analyses were performed using the guidance provided by Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000, and Standard Review Plan (SRP) Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms." The AST analyses include determination of the onsite, specifically the main control room and technical support center, and offsite radiological doses resulting from the HNP limiting design basis accidents (DBAs). The four DBAs considered were the loss-of-coolant accident (LOCA), the fuel handling accident, the control rod drop accident, and the main steam line break accident. The analyses demonstrate that, using AST methodologies, the post-accident onsite and offsite doses remain within regulatory acceptance limits.

As a result of the application of a revised accident source term, the following changes to the Technical Specifications are proposed:

1. The definition of dose equivalent I-131 (DE I-131) is revised to incorporate the updated reference for the dose conversion factors used in the DE I-131 calculation on Units 1 and 2.

2. The maximum allowed reactor coolant specific activity is revised from 4.0 μCi/gm DE I-131 to 2.0 μCi/gm DE I-131 on Units 1 and 2.

3. A Unit 1 Technical Specification on secondary containment bypass leakage is added, consistent with the current licensing basis on Unit 2, and the maximum allowed bypass leakage rate is conservatively increased from 0.9% to 2.0% of the maximum allowable primary containment leakage rate ($L_a$) to allow for newly identified secondary containment bypass leakage paths on both Units 1 and 2.

4. Maximum allowed combined main steam line isolation valve (MSIV) leakage rates are revised by increasing the Unit 1 limit to 100 scfh and decreasing the Unit 2 limit to 100 scfh, and by eliminating the per line leakage rate limit. In addition, two separate surveillance acceptance criteria are provided dependant on leakage rate test pressure. Finally, the requirement to restore MSIV leakage to 11.5 scfh or less following discovery of MSIV leakage not meeting the acceptance criterion has been eliminated.

5. A new Technical Specification for residual heat removal drywell spray is added on Units 1 and 2. Drywell sprays are credited for the reduction of activity in the containment atmosphere as well as pressure and temperature reduction following a LOCA.

6. Changes are being made to the Technical Specification Bases to reflect AST implementation.

**Background**

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

Regulatory Guide 1.183 and SRP Section 15.0.1 were used by SNC in preparing the AST analyses. These documents were prepared by the NRC staff to address the use of ASTs at existing operating power reactors. The regulatory guide establishes the parameters of an acceptable AST and identifies the significant attributes of an AST acceptable to the NRC staff. In this regard, the regulatory guide provides guidance to licensees on acceptable applications for an AST; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on risk; and acceptable radiological analysis assumptions. The SRP provides guidance to the staff on the review of AST submittals.

Acceptance criteria consistent with that required by 10 CFR 50.67 were used to replace the HNP current design basis source term acceptance criteria. The AST analyses were performed for the four limiting DBAs that could potentially result in control room and offsite doses.

The HNP current licensing basis allows for the administration of potassium iodide (KI) to be credited to reduce the 30-day post-accident thyroid radiological dose to the operators in the main control room (MCR) for an interim period of approximately 4 years. This interim licensing basis was requested in preparation for MCR tracer gas inleakage testing, in order to accommodate a range of potential tracer gas inleakage test results. Implementation of the AST will allow the administration of KI in the HNP interim licensing basis to be retired, as well as provide a significant increase in margin for MCR unfiltered inleakage assumed in the radiological dose DBA analyses.

**Justification of Technical Specification Changes**

1. Technical Specification 1.1, Definitions: DOSE EQUIVALENT I-131

   *Current Technical Specification*
   Current Technical Specification 1.1 provides a definition of "Dose Equivalent I-131," and includes references to dose conversion factors listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test

Reactor Sites"; Table E-7 of Regulatory Guide 1.109, Rev. 1, 1977; and ICRP 30, Supplement to Part 1, pages 192-212, Table titled "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity," for calculating thyroid DE I-131.

*Proposed Change*
The current Technical Specification 1.1 definition of "Dose Equivalent I-131" is revised to replace the "thyroid dose" with "committed effective dose equivalent" and to reference only Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988 for the dose conversion factors used in calculating DE I-131.

The proposed revised Technical Specification 1.1 definition is:

> DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same Committed Effective Dose Equivalent as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.

*Justification*
The existing definition is revised to conform to the implementation of AST. The new citation of dose conversion factors is cited in Regulatory Guide 1.183, which has been found to be acceptable by the NRC for AST applications. Inhalation committed effective dose equivalent dose conversion factors used in this calculation are from FGR 11.

With the implementation of AST, the accident dose guidelines of 10 CFR 100 are superseded by the dose criteria of 10 CFR 50.67. The whole body and thyroid doses of 10 CFR 100 are replaced by the total effective dose equivalent (TEDE) criteria of 10 CFR 50.67. A conforming change to the definition is to replace "thyroid dose" in the definition with "committed effective dose equivalent." The analyses performed in support of this amendment request determined radiological consequences in terms of the TEDE dose quantity and were shown to be in compliance with the dose criteria of 10 CFR 50.67. These changes to the definition are acceptable because they reflect adoption of the dose conversion factors and the dose consequences of the revised radiological analyses.

2. Technical Specification 3.4.6, Reactor Coolant System Specific Activity

*Current Technical Specification*
The maximum limit for RCS specific activity in current Technical Specification

3.4.6 Conditions A and B is 4.0 μCi/gm DE I-131.

*Proposed Change*
The maximum limit for RCS specific activity in Technical Specification 3.4.6 Conditions A and B is revised from 4.0 μCi/gm DE I-131 to 2.0 μCi/gm DE I-131.

*Justification*
This revision is required to meet control room dose regulatory limits for the main steam line break accident. A lower maximum allowable RCS specific activity is more conservative than the existing limit. Typical values of DE I-131 RCS specific activity are well below the new limit, and adequate operating margin is maintained.

3. Technical Specification 3.6.1.3, Primary Containment Isolation Valves

*Current Technical Specification*
As stated in surveillance requirement (SR) 3.6.1.3.10 for Unit 2, the maximum combined leakage rate for all secondary containment bypass leakage paths is $0.009L_a$, where $L_a$ is the maximum allowable primary containment leakage rate,. For Unit 1, there is no Technical Specification on maximum leakage rate for secondary containment bypass leakage.

*Proposed Change*
For Unit 2, the current Technical Specification SR is revised to increase the maximum combined leakage rate for all secondary containment bypass leakage paths from $0.009L_a$ to $0.02L_a$. For Unit 1, new Technical Specification SR 3.6.1.3.13 is added. SR 3.6.1.3.13 establishes a maximum combined leakage rate for all secondary containment bypass leakage paths of $0.02L_a$.

*Justification*
The secondary containment bypass leakage rate assumptions in the radiological dose consequence analysis for the LOCA form the basis for the revised Technical Specification limits. The proposed secondary bypass leakage rate limit of $0.02L_a$ is acceptable since this value was assumed in the accident analysis and regulatory criteria have been met. Because calculated doses are below the regulatory limits of 10 CFR 50.67, additional leakage margin exists.

The increase in bypass leakage is necessary to allow for newly identified bypass leakage paths. The addition of this Technical Specification SR to Unit 1 reflects a required Regulatory Guide (RG) 1.183 assumption in the accident analyses and standardizes the Technical Specifications between units.

4. Technical Specification 3.6.1.3, Primary Containment Isolation Valves

*Current Technical Specification*
Unit 1 Technical Specification SR 3.6.1.3.10 specifies a maximum leakage rate of 11.5 scfh through each MSIV when tested at 28.0 psig or greater. Unit 2

Technical Specification SR 3.6.1.3.11 specifies a maximum leakage rate of 100 scfh through each MSIV, and a combined maximum pathway leakage rate of 250 scfh, when tested at 28.8 psig or greater.

Unit 2 Technical Specification SR 3.6.1.3.11 imposes the leakage rate acceptance criteria of 11.5 scfh maximum leakage through an MSIV upon discovery of leakage not meeting the 100 scfh limit.

*Proposed Change*
The per line MSIV leakage rate limits are eliminated from the Technical Specification SR for both units (SR 3.6.1.3.10 for Unit 1, and SR 3.6.1.3.11 for Unit 2). The Unit 1 combined maximum leakage rate is established at 100 scfh when tested at ≥ 28.0 psig and < 50.8 psig. The Unit 2 combined maximum leakage rate is reduced from 250 scfh to 100 scfh when tested at ≥ 28.8 psig and < 47.3 psig. The pressure values of 50.8 psig and 47.3 psig represent calculated peak drywell pressures for Unit 1 and Unit 2, respectively, in the event of a LOCA.

A second test pressure range, with a corresponding leakage rate criterion, is established for both units when test pressure exceeds the peak calculated drywell pressure during a LOCA. This is in addition to the 100 scfh combined maximum leakage rate specification when tested within the specified test pressure range that is below the calculated peak drywell pressure. For Unit 1, a combined maximum leakage rate of 144 scfh is established when tested at ≥ 50.8 psig. For Unit 2, a combined maximum leakage rate of 144 scfh is established when tested at ≥ 47.3 psig.

For Unit 2, the requirement to restore MSIV leakage to 11.5 scfh upon discovery of leakage not meeting the 100 scfh leakage rate limit is eliminated.

*Justification*
The revised values for MSIV combined maximum leakage rates are used in the radiological dose consequence analysis for the LOCA. The contribution to total combined leakage from any individual MSIV is not considered in the analysis. Conservatively, the analysis assumes that the maximum allowed combined leakage rate is entirely through one MSIV. The proposed leakage rates are acceptable since this value was assumed in the revised accident analysis, and calculated doses are below the regulatory criteria of 10 CFR 50.67. In addition, because calculated doses are below the regulatory criteria, additional leakage margin exists.

The addition of a second MSIV leakage rate criterion for testing at or above calculated peak drywell pressure provides a more accurate leakage rate acceptance criterion for test pressures that are higher than calculated post-LOCA peak drywell pressures. This facilitates testing the MSIVs in the accident direction at peak accident drywell pressure as preferred by 10 CFR 50 Appendix J, as opposed to testing the MSIVs in the reverse direction at a lower test pressure as allowed by existing HNP Appendix J exemptions. A higher pressure results in a higher mass flow rate through a given leakage area. The higher leakage rate (mass flow rate) acceptance criterion is based on a pressure and mass flow rate analysis. This allows for the use of different MSIV Appendix J test configurations as dictated by plant configuration during the outage, while also ensuring that the appropriate acceptance criterion exists for the actual test pressure used.

The elimination of the requirement to restore MSIV leakage to 11.5 scfh is acceptable since it is not an input or assumption in the radiological dose consequence analysis. In addition, this restoration is an overly-restrictive maintenance burden. The disadvantages of increased maintenance and higher worker radiation exposure associated with restoring MSIV leakage rates to relatively low values are not justified by any additional conservatism that might apply.

5. Technical Specification 3.6.2.5, Residual Heat Removal Drywell Spray

*Current Technical Specification*
There is currently no Technical Specification for drywell spray.

*Proposed Change*
Technical Specification 3.6.2.5, Residual Heat Removal (RHR) Drywell Spray, is added for both Unit 1 and Unit 2. The Technical Specification, limiting condition for operation (LCO), applicability, action statements, and SRs are patterned after existing Technical Specification 3.6.2.4, RHR Suppression Pool Spray. The proposed LCO requires two RHR drywell spray subsystems to be operable.

*Justification*
The LOCA radiological dose analysis credits the use of drywell sprays for the reduction of airborne activity in the drywell by scrubbing inorganic iodines and particulates from the primary containment atmosphere. In addition, drywell spray is credited in the LOCA analysis for reducing the temperature and pressure in the drywell over time, thereby reducing the post-LOCA primary containment and MSIV leakage to within the assumptions of the dose analysis. Drywell spray is assumed to be manually initiated. Initiation is based on radiation levels in the drywell. By requiring two RHR drywell spray subsystems to be operable, this will ensure that in the event of a design basis LOCA, at least one subsystem will be operable assuming the worst case single active failure.

It should also be noted that the surveillance frequency for new Technical Specification SR 3.6.2.5.2 is "following maintenance which could result in nozzle blockage," unlike the 10 year frequency used for the similar surveillance for RHR suppression pool spray. Given the location of the spray headers in the drywell and prior demonstration of system operability, nozzle blockage is considered unlikely except as a consequence of maintenance or repair. This SR frequency has been approved by the NRC for Perry for containment spray.

6. Technical Specification Bases

*Current Technical Specification Bases*
The Technical Specification Bases provide explanation and rationale for associated Technical Specification requirements, and in some cases, how they are to be implemented.

*Proposed Change*
In addition to the Technical Specification Bases changes associated with the above mentioned Technical Specification changes, other Technical Specification Bases changes are associated with changing the radiological dose limits reference from 10 CFR 100 to 10 CFR 50.67, changing the accident analysis methodology reference to AST RG 1.183, and reflecting the additional standby liquid control system function of buffering the suppression pool to preclude the re-evolution of iodine from the suppression pool water following a DBA LOCA.

*Justification*
The accident dose guidelines of 10 CFR 100 are superseded by the dose criteria of 10 CFR 50.67. The whole body and thyroid doses of 10 CFR 100 are replaced by the TEDE criteria 10 CFR 50.67, and references to 10 CFR 100 are replaced with 10 CFR 50.67. This is a conforming change.

The buffering of the suppression pool is credited in the radiological dose analysis for the LOCA and the resultant calculated doses are below the regulatory criteria of 10 CFR 50.67. The remaining Bases changes reflect the use of the AST RG 1.183 accident analysis methodology.

Other changes were made to the Technical Specification Bases to conform to the changes being made to the associated Technical Specifications. The revisions to the Technical Specification Bases incorporate supporting information for the proposed Technical Specification changes. Bases do not establish actual requirements, and as such do not change technical requirements of the Technical Specifications. The Bases changes are therefore acceptable, since they administratively document the reasons and provide additional understanding for the associated Technical Specification requirements.

Enclosure 3

Edwin I. Hatch Nuclear Plant
Request to Implement an Alternative Source Term

10 CFR 50.92 Significant Hazards Evaluation and Environmental Assessment

## 10 CFR 50.92 Significant Hazards Evaluation and Environmental Assessment

### Proposed Change

Southern Nuclear Operating Company (SNC) requests a revision of the Edwin I. Hatch Nuclear Plant (HNP) Operating License by revising the Technical Specifications and incorporating an alternative source term (AST) methodology into the facility's licensing basis. The proposed license amendment involves a full scope implementation of an AST methodology by revising the current accident source term and replacing it with an accident source term as prescribed in 10 CFR 50.67.

AST analyses were performed using the guidance provided by Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000, and Standard Review Plan, Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms." The AST analyses include determination of the onsite, specifically the main control room and technical support center, and offsite radiological doses resulting from the HNP limiting design basis accidents (DBAs). The four DBAs considered were the loss-of-coolant accident (LOCA), the fuel handling accident, the control rod drop accident, and the main steam line break accident. The analyses demonstrate that, using AST methodologies, the post-accident onsite and offsite doses remain within regulatory acceptance limits.

As a result of the application of a revised accident source term, changes to the Technical Specifications are proposed that revise the definition of dose equivalent I-131, revise the maximum allowed reactor coolant specific activity, revise secondary containment bypass leakage rates, revise main steam line isolation valve leakage rates, and add a requirement for residual heat removal drywell sprays. Changes are also proposed to Technical Specification Bases to reflect AST implementation.

In addition to Technical Specification changes, the AST implementation adds a post-LOCA suppression pool pH control function for the standby liquid control system, credits main steam piping and the main condenser to provide an alternate leakage treatment pathway for main steam isolation valve leakage for Unit 1 (previously approved for Unit 2), and credits the turbine building ventilation system for removing activity from the turbine building post-accident.

### 10 CFR 50.92 Evaluation

In 10 CFR 50.92(c) the Nuclear Regulatory Commission (NRC) provides the following standards to be used in determining the existence of a significant hazards consideration:

...a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22, or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in the margin of safety.

SNC has reviewed the proposed amendment request and determined that its adoption does not involve a significant hazards consideration based upon the following discussion:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Adoption of the AST and those plant systems affected by implementing AST do not initiate DBAs. The AST does not affect the design or manner in which the facility is operated; rather, once the occurrence of an accident has been postulated, the new accident source term is an input to analyses that evaluate the radiological consequences. The implementation of the AST and changed Technical Specifications have been incorporated in the analyses for the limiting DBAs at HNP.

   The structures, systems, and components affected by the proposed change are mitigative in nature, and relied upon after an accident has been initiated. Based on the revised analyses, the proposed changes to the Technical Specifications (including revised leakage limits) impose certain performance criteria which do not increase accident initiation probability. The proposed changes do not involve a revision to the parameters or conditions that could contribute to the initiation of a DBA discussed in Chapter 15 of the Unit 2 Final Safety Analysis Report. Therefore, the proposed change does not result in an increase in the probability of an accident previously identified.

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

2. **Does the proposed change create the possibility of a new or different kind of accident from any previously evaluated?**

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

3. **Does the proposed change involve a significant decrease in the margin of safety?**

   The changes proposed are associated with a revision to the licensing basis for HNP. Approval of the licensing basis change from the original source term to the AST is requested by this application for a license amendment. The results of the accident analyses revised in support of the proposed change are subject to the acceptance criteria in 10 CFR 50.67. The analyzed events have been carefully selected, and the analyses supporting these changes have been performed using approved methodologies and conservative inputs to ensure that analyzed events are bounding and safety margin has been retained. The dose consequences of these limiting events are within the acceptance criteria presented in 10 CFR 50.67, Regulatory Guide 1.183, and Standard Review Plan 15.0.1. Therefore, because the proposed changes continue to result in dose consequences within the applicable regulatory limits, the changes are considered to not result in a significant reduction in the margin of safety.

**Environmental Assessment**

SNC has evaluated the proposed changes and determined the changes do not involve (1) a significant hazards consideration, (2) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (3) a significant increase in the individual or cumulative occupational exposure. Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9), and an environmental assessment of the proposed changes is not required.

Enclosure 4

Edwin I. Hatch Nuclear Plant
Request to Implement an Alternative Source Term

Regulatory Safety Analysis

Enclosure 4
Edwin I. Hatch Nuclear Plant
Request to Implement an Alternative Source Term

## Regulatory Safety Analysis

Southern Nuclear Operating Company (SNC) requests a revision of the Edwin I. Hatch Nuclear Plant (HNP) Operating Licenses to revise the Technical Specifications and incorporate an alternative source term (AST) methodology into the facility's licensing basis. The proposed license amendment involves a full scope implementation of an AST methodology by revising the current accident source term and replacing it with an accident source term as prescribed in 10 CFR 50.67.

The AST analyses include determination of the onsite, specifically the main control room and technical support center, and offsite radiological doses resulting from the HNP limiting design basis accidents (DBAs). The four DBAs considered were the loss-of-coolant accident, the fuel handling accident, the control rod drop accident, and the main steam line break accident. The analyses demonstrate that, using AST methodologies, the post-accident onsite and offsite doses remain within regulatory acceptance limits.

The application of a revised accident source term and the application of an AST methodology have resulted in several proposed changes to the Technical Specifications, as well as changes to inputs and assumptions in the current design basis analyses.

The proposed license amendment will comply with 10 CFR 50.67. The radiological dose consequence analyses have shown that the dose criterion, 25 rem total effective dose equivalent (TEDE), prescribed for (1) an individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, and (2) an individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), has been met. Additionally, the analyses have demonstrated that adequate radiation protection would be provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident.

The proposed license amendment will comply with 10 CFR 50 Appendix A General Design Criterion (GDC) 19. GDC 19 requires maintaining the control room in a safe, habitable condition under accident conditions, including loss-of-coolant accidents. Radiological dose consequence analyses have demonstrated that adequate radiation protection is provided to permit access and occupancy of the control room under postulated accident conditions, and that the radiation exposures would not exceed 5 rem TEDE for the duration of the accident.

The proposed license amendment will conform to Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000. RG 1.183 provides regulatory guidance on acceptable applications of ASTs; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. This guide establishes an acceptable AST and identifies the significant attributes of an AST that may be found acceptable by the NRC staff. RG 1.183 also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

The analyzed events have been carefully selected. The analyses supporting the proposed changes have been performed using approved methodologies and conservative inputs to ensure that analyzed events are bounding and safety margin has been retained. The dose consequences of these limiting events are within the acceptance criteria presented in 10 CFR 50.67, RG 1.183, and Standard Review Plan Section 15.0.1.

Therefore, the proposed changes to Technical Specifications and a full scope implementation of an AST methodology contained within the license amendment request comply with the applicable regulatory requirements and guidance.

In conclusion, based on the considerations discussed above, (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) issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.