March 14, 2003

Mr. Paul D. Hinnenkamp
Vice President - Operations
Entergy Operations, Inc.
River Bend Station
P.O. Box 220
St. Francisville, LA 70775

SUBJECT: RIVER BEND STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: FULL-SCOPE APPLICATION OF ALTERNATIVE SOURCE TERM INSIGHTS (TAC NO. MB5021)

Dear Mr. Hinnenkamp:

The U. S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 132 to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1. The amendment consists of changes to the Technical Specifications in response to your application dated April 24, 2002, as supplemented by letters dated July 18, December 18 and 20, 2002, and February 19, 2003.

The amendment reflects a full-scope implementation of the alternative source term, as described in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," pursuant to 10 CFR 50.67, "Accident source term."

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

Sincerely,

/RA/

Michael Webb, Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosures: 1. Amendment No. 132 to NPF-47
2. Safety Evaluation

cc w/encls: See next page

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ENTERGY GULF STATES, INC. **

AND

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-458

RIVER BEND STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 132
License No. NPF-47

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

   A. The application for amendment by Entergy Gulf States, Inc.* (the licensee) dated April 24, 2002, as supplemented by letters dated July 18, December 18 and 20, 2002, and February 19, 2003, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

* Entergy Operations, Inc. is authorized to act as agent for Entergy Gulf States, Inc., and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

**Entergy Gulf States, Inc., has merged with a wholly owned subsidiary of Entergy Corporation. Entergy Gulf States, Inc., was the surviving company in the merger.

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-47 is hereby amended to read as follows:

   (2) <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 132 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. EOI shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Robert A. Gramm, Chief, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 14, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 132

FACILITY OPERATING LICENSE NO. NPF-47

DOCKET NO. 50-458

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

Remove	Insert
1.0-2	1.0-2
3.3-71	3.3-71
3.6-7	3.6-7
3.6-8	3.6-8
3.6-18	3.6-18
3.6-19	3.6-19
3.6-20	3.6-20
3.6-47	3.6-47
3.6-60	3.6-60
3.6-66	3.6-66
3.7-5	3.7-5
3.7-6	3.7-6
3.7-7	3.7-7
3.7-9	3.7-9
3.7-10	3.7-10
3.7-11	3.7-11
3.8-17	3.8-17
3.8-18	3.8-18
3.8-19	3.8-19
3.8-28	3.8-28
3.8-29	3.8-29
3.8-36	3.8-36
3.8-41	3.8-41
5.0-12	5.0-12
5.0-16	5.0-16

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 132 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-47

ENTERGY OPERATIONS, INC.

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

1.0 INTRODUCTION

By application to the U. S. Nuclear Regulatory Commission (Commission, NRC, the staff) dated April 24, 2002, as supplemented by letters dated July 18, December 18 and 20, 2002, and February 19, 2003, Entergy Operations, Inc. (the licensee), requested changes to the Technical Specifications (TSs) for the River Bend Station, Unit 1 (RBS). The supplements dated July 18, December 18 and 20, 2002, and February 19, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 11, 2002 (67 FR 40021).

The proposed changes would facilitate full-scope implementation of the alternative source term (AST), as described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," pursuant to Section 50.67 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.67), "Accident source term." The current RBS accident source term was developed using U.S. Atomic Energy Commission Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," published in 1962.

Specifically, the licensee requested that:

1. TS Section 1.1, "Definitions," be amended to re-define "Dose Equivalent I-131" and delete reference to TID-14844.

2. TS Section 3.3.7.1, "Control Room Fresh Air (CRFA) System Instrumentation" be amended to revise Note (b) in Table 3.3.7.1-1 to reflect that the requirements are only applicable during movement of "recently irradiated fuel."

3. TS Section 3.6.1.2, "Primary Containment Air Locks" be amended to revise Surveillance Requirement (SR) 3.6.1.2.1 to remove requirements for annulus bypass leakage and revise SR 3.6.1.2.4 to increase the leakage rate for the air lock seal pneumatic system from 1.28 psig per day to 1.5 psig per day.

4. TS Section 3.6.1.3, “Primary Containment Isolation Valves,” be amended to revise SR 3.6.1.3.9 and SR 3.6.1.3.10 and to delete SR 3.6.1.3.12.

5. TS Section 3.6.4.1, “ Secondary Containment - Operating,” be amended to revise SR 3.6.4.1.4 to increase the Auxiliary Building drawdown time from 13.5 to 34.5 seconds.

6. TS Section 3.6.5.1, “Drywell,” and “Drywell Air Locks,” be amended to revise SR 3.6.5.1.2 and SR 3.6.5.2.5 to increase the leakage rate for the air lock seal pneumatic system from 0.67 psig per day to 20.0 psig per day.

7. TS Section 3.7.2, “Control Room Fresh Air (CRFA) System,” be amended to revise its APPLICABILITY, CONDITION C and its associated REQUIRED ACTIONS, and CONDITION E and its associated REQUIRED ACTIONS.

8. TS Section 3.7.3, “Control Room AC System,” be amended to revise its APPLICABILITY, CONDITION D and its associated REQUIRED ACTIONS, and CONDITION E and its associated REQUIRED ACTIONS.

9. TS Section 3.8.2, “AC Sources - Shutdown,” be amended to revise its APPLICABILITY and REQUIRED ACTIONS associated with CONDITION A and CONDITION B.

10. TS Section 3.8.5, “DC Sources - Shutdown,” be amended to revise its APPLICABILITY and REQUIRED ACTIONS associated with CONDITION A.

11. TS Section 3.8.8, “Inverters - Shutdown,” be amended to revise its APPLICABILITY and REQUIRED ACTIONS associated with CONDITION A.

12. TS Section 3.8.10, “Distribution Systems - Shutdown,” be amended to revise its APPLICABILITY and REQUIRED ACTIONS associated with CONDITION A.

13. TS Section 5.5.7, “Ventilation Filter Testing Program (VFTP),” be amended to revise the Standby Gas Treatment System (SGTS) allowable penetration from 0.5 percent to 5.0 percent and Control Room Fresh Air (CRFA) System allowable penetration from 0.5 percent to 1.0 percent.

14. TS Section 5.5.13, “Primary Containment Leakage Rate Testing Program,” be amended to increase the containment leakage rate from 0.26 percent per day to 0.325 percent per day.

For these proposed TS changes, the licensee re-analyzed and submitted the radiological consequences for four affected design-basis accidents (DBAs): the loss-of-coolant accident (LOCA), the fuel handling accident (FHA), the control rod drop accident (CRDA), and main steam line break (MSLB) accident.

2.0 REGULATORY EVALUATION

Since the publication of TID-14844, significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. The current radiological consequence analyses for the DBAs for RBS are based upon the TID-14844 accident source term. In 1995, the NRC staff published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." NUREG-1465 provided estimates of the accident source term that were more physically based and that could be applied to the design of future light-water power reactors. NUREG-1465 presents a representative AST for a boiling-water reactor (BWR) and for a pressurized-water reactor (PWR). These source terms are characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release to the containment.

The staff considered the applicability of the revised source terms in NUREG-1465 to operating reactors and determined that the current analytical approach based on the TID-14844 source term would continue to be adequate to protect public health and safety. Operating reactors licensed under that approach would not be required to re-analyze accidents using the revised source terms. The staff also determined that some licensees might wish to use an AST in their analyses to support cost-beneficial licensing actions. The staff, therefore, initiated several actions to provide a regulatory basis for operating reactors to use an AST in design basis radiological consequence analyses. The results and findings of an evaluation of the impact of implementing the AST for operating reactors are presented in SECY-98-154, “Results of the Revised Source Term Rebaselining for Operating Reactors.”

The Commission approved the use of the alternative source term at operating reactors in Staff Requirements Memorandum 99-240, dated December 8, 1999, stating that, “This action would allow interested licensees to pursue cost-benefit licensing actions to reduce unnecessary regulatory burden without compromising the safety of the facility. Many of the alternative source term applications may provide concurrent improvements in overall safety and in reduced occupational exposures.”

These initiatives resulted in the development and issuance of 10 CFR 50.67 and RG 1.183. As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67 replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR 50, Appendix A, General Design Criterion (GDC) -19, “Control Room,” as follows:

	10 CFR 50.67 GDC-19	10 CFR 100.11 GDC-19
Exclusion Area Boundary and Low Population Zone	25 rem TEDE	300 rem thyroid and 5 rem whole body
Control Room	5 rem TEDE	5 rem whole body, or its equivalent to any part of the body

A holder of an operating license issued prior to January 10, 1997, or a holder of a renewed license under 10 CFR Part 54 whose initial operating license was issued prior to January 10, 1997, is allowed by 10 CFR 50.67 to voluntarily revise its current accident source term used in design basis radiological consequence analyses for a license amendment under 10 CFR 50.90, “Application for amendment of license or construction permit." In this license amendment, the licensee requested a full-scope implementation of the AST, as described in RG 1.183 pursuant to 10 CFR 50.67. In general, information provided by RG 1.183 is reflected in Chapter 15.0.1 of the Standard Review Plan (SRP), “Radiological Consequence Analyses Using Alternative Source Terms.”

Other relevant regulatory requirements applicable to this license amendment are: (1) GDC-19, and (2) NUREG-0737 III.D.3.4 as it relates to maintaining the control room in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases.

3.0 TECHNICAL EVALUATION

The NRC staff has reviewed the licensee's regulatory and technical analyses in support of its proposed license amendment which are described in Attachments 1, 5, 6, 7, 8, 9, and 10 of the licensee's submittal dated April 24, 2002; Attachments 1, 5, 6, 7, 8, 9, and 10 of the licensee's submittal dated December 18, 2002; and Attachment 1 of the licensee's submittal dated December 20, 2002. The detailed evaluation below will support the conclusion that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

The licensee re-analyzed and submitted the radiological consequence analyses for the following four DBAs:

(1) LOCA
(2) FHA
(3) CRDA
(4) MSLB Accident

The NRC staff's evaluation is based on the licensee's submittals through December 2002. The staff reviewed the licensee's submittal dated February 19, 2003, and found that information provided in that submittal concerning the revised atmospheric relative concentrations (χ/Q values) at the control room air intake are not significantly different or are less restrictive than the earlier submittals. Therefore, the NRC staff has not incorporated the revised χ/Q values in its radiological consequence evaluation.

3.1 Loss-of-Coolant Accident (LOCA)

The current radiological consequence analysis for the postulated LOCA using the TID-14844 source term is provided in the RBS Updated Safety Analysis Report (USAR) Section 15.6. To demonstrate that the RBS engineered safety features (ESFs) designed to mitigate the radiological consequences will remain adequate after this license amendment, the licensee re-evaluated the offsite and control room radiological consequences of the postulated LOCA. The licensee has implemented the AST in this re-evaluation. The licensee submitted the results of its offsite and control room dose calculations (see Table 1).

In addition, the licensee provided a complete dose analysis and described the major assumptions and parameters used in its dose calculations. The licensee also provided the fission product transport, removal, and release models developed and used for this license amendment request. In its dose calculations, the licensee used the RADTRAD computer code described in NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation." The code was developed by an NRC contractor for the NRC staff. The code estimates transport and removal of fission products and dose at selected receptors.

As documented in the submittals, the licensee has performed accident analyses using the AST; and, based on these analyses, concluded that the existing ESF systems at RBS will provide assurance that the total radiological consequences of the postulated LOCA at the exclusion area boundary (EAB), in the low population zone (LPZ), and in the control room meet the radiation dose criteria specified in 10 CFR 50.67. The NRC staff has reviewed the licensee's analysis and has performed an independent confirmatory radiological consequence dose calculation for the following three potential fission product release pathways after the postulated LOCA:

(1) main steam isolation valve (MSIV) and secondary containment bypass leakages,
(2) containment leakage, and
(3) post-LOCA leakage from ESF systems outside containment.

3.1.1 Main Steam Isolation Valve (MSIV) and Secondary Containment Bypass Leakage Pathways

There are four main steam lines (MSLs) at RBS. Each MSL has an inboard MSIV, an outboard MSIV, and a main steam shutoff valve. These valves isolate the reactor coolant system (RCS) in the event of a break in a steam line outside the primary containment, a design basis LOCA, or any other event requiring containment isolation.

Although the MSIVs are designed to provide a leak-tight barrier, it is recognized that some leakage occurs through these valves. The RBS MSIV leakage control system (LCS) minimizes potential release of fission products from the RCS through the closed MSIVs after a postulated design basis LOCA. The system is composed of two independent inboard and outboard systems. The leakage control barrier is established by pressurizing the isolated volumes in the MSLs between the inboard and outboard isolation valves maintaining a pressure of at least 10 percent over the prevailing post-LOCA reactor vessel pressure. The pressurized volume eliminates out-leakage through the closed MSIVs such that any leakage that does occur is inward from the pressurized volume into the reactor pressure vessel or containment.

The current RBS TS limit for MSIV leakage is less than or equal to 150 standard cubic feet per hour (scfh) through all four MSLs. The licensee conservatively assumed in its analysis that one of the two MSIVs in one MSL fails to close (single active failure) resulting in 50 scfh leakage and it is released directly to the environment through the remaining MSIV in this MSL for 25 minutes prior to the MSIV-LCS becoming fully operational after the accident. RBS currently has an administrative limit of 30 scfh per MSIV. The leakage is assumed to originate from the drywell bypassing the containment and released directly to the environment via the turbine building. After 25 minutes from the initiation of the postulated LOCA, no leakage through MSLs is assumed.

In its safety evaluation in NUREG-0989, "Safety Evaluation Report Related to the Operation of RBS (May 1984)," the NRC staff assumed no MSIV failure to close and, therefore, no leakage was assumed. The licensee also did not assume MSIV failure in its current RBS USAR and the MSLs were not considered as a leakage path. The staff finds the failure of one of the two MSIVs in one MSL to close resulting in 50 scfh leakage to be conservative and, therefore, the licensee's assumption is acceptable. No credit is provided for fission product deposition or holdup for decay in this steam line.

The licensee also assumed secondary containment bypass (SCB) leakage at the proposed TS limit of 5.8E+5 cubic centimeters per hour (cc/hr), or 0.341 percent containment air volume per day, for the first 24 hours of the postulated LOCA. Since containment pressure is the driving force for this leakage pathway, the licensee assumed that SCB is also reduced to 55 percent of the original value after 24 hours based on the containment pressure response (as the licensee assumed for the containment leak). The current TS limit is 1.7E+5 cc/hr (TS Section 3.6.1.3.9). The SCB leakage is from the primary containment building which will bypass the annulus and auxiliary building and be released directly to the environment without filtration. This leakage is independent of the overall containment leak rate summation. The licensee selected and determined that the proposed SCB leakage value meets the dose criterion set forth in 10 CFR 50.67. The SCB pathway contributed the most radiological consequence dose at the EAB, the LPZ and control room (greater than 68 percent).

3.1.1.1 Fission Product Transport in Drywell

The licensee assumed, and the NRC staff agrees, that a large-break LOCA as a result of a double-ended guillotine pipe rupture would be the most limiting LOCA with respect to the offsite and control room radiological consequences. The break releases reactor coolant to the drywell. No water injection from the emergency core cooling system (ECCS) is assumed and the reactor water level drops below the core, exposing the reactor fuel. In Grand Gulf Nuclear Station (Grand Gulf) License Amendment No. 143, issued on March 22, 2000, "Implementation of Alternative Source Term Limited Scope Application for the Timing of the Onset of Gap Activity Release," the staff evaluated the earliest time of fission products release (fuel gap activity release) from perforated fuel rods following a postulated LOCA and concluded that the minimum time would be no earlier than 120 seconds. This finding is also applicable to the RBS design as RBS and Grand Gulf are designed with the same BWR Mark III containment.

The NRC staff also assumed that all fission products are released directly to the drywell and leaked into the primary containment and into the main steam lines, bypassing the suppression pool. The staff concludes that these assumptions are appropriate for the large-break LOCA. For small-break LOCAs with operator actuation of an automatic depressurization system (ADS), most of the fission products would be released into the drywell through the pipe break and into the suppression pool through the ADS, where the fission products are removed.

As characterized in NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” the gap and early in-vessel fission product releases terminate 2 hours after the postulated LOCA initiation. The NRC staff assumed [as it did for Perry Nuclear Station (Perry)] License Amendment No.103, issued on March 26, 1999, “Main Steam Line Leakage Requirements and Elimination of the Main Steam Isolation Valve Leakage Control System Implementing the Alternative Source Term,” and for Grand Gulf License Amendment No.145, issued on March 14, 2001, “Full-Scope Implementation of an Alternative Accident Source Term”) that the fission products are homogeneously distributed between the drywell and the primary containment 2 hours after accident initiation. This would require reflooding of the reactor vessel. Instead of trying to justify an all encompassing steaming rate due to this reflooding, the staff concludes that a substantial amount of fission products may end up in the primary containment as well as the drywell. For most of the risk significant cases, such as station blackout and transients, all the fission products are released directly to the primary containment via the safety relief valves. Waiting 2 hours to homogeneously mix the source term is acceptable for achieving an appropriate balance because the worst 2 hours are considered, not the first 2 hours used with the TID-14844 source term.

Confirmatory calculations performed by the NRC staff showed that the radiological consequences are dependent upon the drywell bypass leakage prior to the termination of fission product release at 2 hours. Because of this sensitivity, the staff concludes, as it did for Perry and Grand Gulf, that, without relocation to the lower head, the steaming rate of an intact core on the order of 3,000 cubic feet per minute (cfm) should be assumed for drywell bypass leakage. The staff’s steaming rate prior to 2 hours is conservative in that it does not credit steaming due to relocation, cooling from alternative water sources, or the release of hydrogen gas.

The 3,000 cfm drywell bypass leakage rate is based upon large-break LOCA analyses performed with MELCOR on a Grand Gulf type model. These analyses showed no relocation below the core plate, water level below the core plate, and an average steaming rate of approximately 2,800 cfm prior to quenching of the core at approximately 0.5 hours. Also, alternative water sources, such as the standby liquid control system, would not be available during station blackout sequences, which comprised 96 percent of the core damage frequency in the Grand Gulf case. Therefore, the NRC staff concludes the use of 3,000 cfm for the drywell bypass leakage prior to 2 hours is also reasonable for RBS.

3.1.1.2 Aerosol Deposition Within the Drywell

In its evaluation, the NRC staff used a simplified model developed by the staff’s contractor for estimating the fission product aerosol deposition by natural processes in the drywell of BWRs following a postulated LOCA. The model is described in NUREG/CR-6189, “A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containment.” This model was derived by correlating the results of Monte Carlo uncertainty sampling analyses and assessing uncertainties in aerosol properties, drywell geometries, accident progression, and aerosol behavior expected to be associated with a postulated LOCA in the drywell.

The NRC staff assumed that the fission product aerosols in the drywell are removed by natural processes (gravitational sedimentation and phoretic phenomena such as diffusiophoresis and thermophoresis). The staff assumed that the drywell is well mixed during the entire duration of the accident. The aerosol removal rates used by the staff represent the 90th percentile of the uncertainty distributions (see Table 2). For the main steam lines, the licensee did not request and the staff has not provided any credit for aerosol deposition.

3.1.2 Containment Leakage Pathway

The primary containment consists of a drywell, a wetwell, and supporting systems to limit fission product leakage during and following the postulated LOCA with rapid isolation of the containment boundary penetrations. The current maximum allowable primary containment leakage rate (L_a) is 0.26 percent of primary containment air weight per day. In this amendment request, the licensee proposed (and the NRC staff used in its evaluation) 0.325 percent per day.

In RG 1.183, the NRC staff stated that these leak rates may be reduced 24 hours into the postulated LOCA if supported by plant configuration and analyses to a value not less than 50 percent of the TS leak rate limit. The licensee proposed the containment leak rate be reduced to 55 percent of that value at 24 hours. The staff reviewed the licensee's submittal and accepted the leakage reduction at 24 hours as it did for Grand Gulf.

The RBS secondary containment (which surrounds the primary containment) will collect and retain any fission product leakage from the primary containment and will release fission products to the environment through the standby gas treatment system (SGTS) following the postulated LOCA. The licensee assumed, for the first 30 minutes of the postulated LOCA, the secondary containment will be above a pressure of 0.25 inch water gauge. Therefore, the licensee assumed, and the NRC staff agrees, that the entire primary containment leakage is released directly to the environment during the first 30 minutes of the postulated LOCA. After 30 minutes, the SGTS draws 2500 cfm of secondary containment atmosphere air through a high efficiency particulate air (HEPA) filter with a 99 percent aerosol removal efficiency and a charcoal adsorber with a 90 percent iodine removal efficiency before release to the environment. RBS does not have containment sprays.

3.1.3 Post-LOCA Leakage Pathway From Engineered Safety Features Outside Containment

Any water leakage from ESF components located outside the primary containment releases fission products during the recirculating phase of long-term core cooling following a postulated LOCA. The licensee calculated this leakage to be less than 1 gallon per minute (gpm) and assumed ESF leakage to begin at the time of the postulated LOCA through the entire duration of the accident (30 days). The use of a 1 gpm leakage rate is a departure from the guidance provided in RG 1.183, which states that the leakage should be taken as two times the design leakage rate.

The licensee stated that the 1 gpm leakage value is contained in the original RBS USAR which was approved by the NRC staff. Therefore, the licensee considers that this assumption is a part of the RBS original license and that it is still a part of the current licensing basis. The staff accepted this leakage value as a design basis in its NUREG-989, "Safety Evaluation Report Related to the Operation of RBS (May 1984)" at the time of the initial licensing of RBS. Subsequently, the staff also used this leakage value in RBS License Amendment Nos. 98, 113, and 114. Therefore, the staff finds that the leakage rate used by the licensee is acceptable. The licensee also assumed that 10 percent of all forms of iodine contained in the leakage is released directly to the environment consistent with the guidelines provided in RG 1.183.

3.1.4 Resulting Radiological Consequences from the Postulated LOCA

The licensee re-evaluated the radiological consequences resulting from the postulated LOCA using the AST and concluded that the radiological consequences at the EAB, LPZ and in the control room are within the dose criteria specified in 10 CFR 50.67. The NRC staff has reviewed the licensee's re-evaluation. In performing this review, the staff relied upon information provided by the licensee, staff experience in performing similar reviews, and, where deemed necessary, on staff's confirmatory calculation.

To verify the licensee's radiological consequence analyses, the NRC staff performed its confirmatory radiological consequence dose calculation and found its results are also within the dose criteria specified in 10 CFR 50.67. Although the staff performed its independent radiological consequence dose calculation as a means of confirming the licensee's results, the staff's acceptance is based on the licensee's analyses. The results of the licensee's radiological consequence dose calculation are provided in Table 1 and the major parameters and assumptions used by the licensee and the staff are listed in Tables 2 through 9. The radiological consequences calculated by the licensee and by the staff for the EAB and at the LPZ, and in the control room are all within the dose criteria specified in 10 CFR 50.67. Therefore, the staff concludes that the proposed TS changes implementing the AST meet the relevant dose acceptance criteria.

3.1.5 Post-LOCA Suppression Pool pH Evaluation

The NRC staff reviewed the portion of the submittal dealing with the methodology for maintaining the suppression pool pH above 7 for the 30 day period after a LOCA. It also reviewed the material in the letter dated July 18, 2002 containing the responses to the staff's request for additional information.

There is no special provision in boiling water reactors (BWRs) for controlling the suppression pool pH during plant operation. After a LOCA, its value will depend, therefore, on the chemical species dissolved in the suppression pool water. These species could be released from the damaged core, generated in the radiation fields existing in the containment and drywell after a LOCA, or, in the case of sodium pentaborate, release from the standby liquid control system via the reactor vessel. Most of the chemical species introduced into the suppression pool are either acidic or basic and the resultant suppression pool pH will depend on their relative concentrations and on the buffering action of the sodium pentaborate added to the suppression pool water.

3.1.5.1 Chemical Species Dissolved in Suppression Pool

Cesium and iodine are the two fission products released from the damaged core and dissolved in the suppression pool water. Some of the dissolved cesium is in a form of cesium hydroxide and is a source of OH^- ions. Some of the dissolved iodine remains in a form of hydriodic acid which is a strong acid providing H^+ ions. The nitric and hydrochloric acids are the chemical species produced in the radiation environment existing in the containment after a LOCA.

3.1.5.1.1 Nitric Acid

The amount of nitric acid generated by irradiation of water and air in the containment after a LOCA was determined using the methodology described in NUREG/CR-5950, "Iodine Evolution and pH Control." Its production is proportional to the time-integrated radiation dose rate for gamma and beta radiation. The licensee calculated that 497 gm-moles of nitric acid are generated over the 30 day transient.

3.1.5.1.2 Hydrochloric Acid

Hydrochloric acid is generated in the post-LOCA environment by radiolytic decomposition of the Hypalon cable jacketing by beta and gamma radiation. Its amount is proportional to the radiation energy absorbed by the jacketing. The production of hydrochloric acid was determined by the methodology described in NUREG/CR-5950 and NUREG-1081, "Post-Accident Gas Generation from Radiolysis of Organic Materials." About 38,000 lbs. of Hypalon cable jacketing existing in the plant produce 899 gm-moles of hydrochloric acid.

3.1.5.1.3 Sodium Pentaborate

The primary objective of sodium pentaborate is to control reactivity in the core and it is injected into the reactor vessel by the standby liquid control system. However, it leaks out through the break and gets into the suppression pool water where it produces a buffering action. The operators are directed to manually initiate injection of sodium pentaborate solution upon initiation of severe accident procedures. In the analysis of the suppression pool pH, however, the licensee made a conservative assumption that the injection is initiated 2 hours after the postulated accident and 1657 gallons of the 7.13 percent solution of sodium pentaborate is injected into the reactor vessel at a rate of 41.2 gpm.

3.1.5.2 Determination of Suppression Pool pH

The methodology used for determining pH in the suppression pool was based on a guidance in NUREG/CR-5950. The licensee determined the change of the suppression pool pH during the 30-day transient for two cases. In the first case, no buffering action of sodium pentaborate was assumed, but credit was taken for cesium hydroxide. In this case pH of the suppression pool water dropped below 7 before the end of the 30 day transient. In the second case, buffering action of sodium pentaborate was assumed, but no credit was taken for the presence of cesium hydroxide. In this case the value pH stayed above 7 for the duration of the 30 day transient.

In summary, the licensee described its methodology used for determining suppression pool pH during 30-day transient. The pH of the suppression pool water without addition of sodium pentaborate was determined by relative amounts of the acidic and basic chemicals dissolved in the pool's water. Addition of sodium pentaborate produced buffering action and the decrease of pH as more acidic chemicals were added was less pronounced. The licensee has demonstrated that without sodium pentaborate it was not possible to maintain suppression pH above 7 for the whole duration of the 30-day transient. However, introduction of 1657 gallons of 7.13 percent sodium pentaborate solution will produce sufficiently strong buffering action to ensure that pH will stay above 7 for the whole duration of the 30-day transient.

The NRC staff reviewed the licensee's methodology and performed its independent verification. On the basis of this evaluation, the staff concludes that the post-LOCA suppression pool pH, specified in Attachment 5 of the licensee's submittal, represents a realistic estimate of the pH which would exist in the suppression pool for 30 days after a LOCA. It also concurs with the licensee's findings that with addition of the sufficient amount of sodium pentaborate it is possible to maintain pH above 7 during the post-LOCA transient.

3.2 Fuel Handling Accident (FHA)

The current radiological consequence analysis for the postulated design basis FHA is based on the accident source term described in TID-14844 and it is provided in RBS USAR Section 15.7.4. The licensee re-evaluated the radiological consequences of a postulated FHA in the containment with no credit taken for containment isolation implementing the AST. The FHA is postulated to occur as a consequence of a failure of the fuel assembly lifting mechanism, resulting in a drop of a raised fuel assembly onto stored fuel assemblies in either the spent fuel pool or the reactor core. The licensee stated that all fuel types used by RBS were evaluated and determined General Electric (GE) fuel type 11 (9X9 array) to be bounding.

The licensee assumed a total of 150 GE 9X9 fuel rods are damaged. Instantaneous release of all noble gases and iodine vapors from the fuel rod gaps from the damaged fuel rods occurs as gas bubbles up through the water covering the fuel. All fission products reaching the building atmosphere (either containment or fuel handling building) are released directly to the environment within 2 hours without filtration. Since the assumptions and parameters used for a FHA inside containment are identical to those for an FHA in the fuel handling building, the resulting radiological consequences are the same regardless of the location of the accident.

The licensee concluded in the submittals that the radiological consequences resulting from the postulated FHA in the containment with no credit taken for containment isolation are within the dose acceptance criteria specified in Standard Review Plan (SRP) 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," and GDC-19.

The licensee reached this conclusion as a result of:

(1) implementing the AST,
(2) taking no credit for containment isolation,
(3) taking no credit for iodine removal by the main control room and fuel building charcoal adsorbers,
(4) using an overall decontamination factor of 200 for iodine in elemental and particulate forms in the spent fuel pool water with minimum water depth of 23 feet consistent with the guidelines provided in RG 1.183,
(5) releasing all fission products within 2 hours using an exponential release model with higher release in the initial period,
(6) assuming all fuel rods in one fuel assembly with an axial power peaking factor of 2.0 are damaged to the extent that its entire gap activity inventory of the damaged fuel rods is released to the surrounding water,
(7) using a fission product decay period of 24 hours (time period from the reactor shutdown to the first fuel movement),

(8) taking credit for manual selection of the more favorable air intakes by control room operators in accordance with the guidance provided in SRP Section 6.4 (see Section 3.6.3), and
(9) using the guidance provided in Appendix B to RG 1.183, "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident."

The NRC staff reviewed the methods, parameters, and assumptions used by the licensee in its radiological dose consequence analyses and finds that they are consistent with the conservative guidance provided in RG 1.183. To verify the licensee's radiological consequence assessments, the staff performed confirmatory radiological consequence dose calculations for the postulated FHA. The radiological consequences calculated by the staff are within the dose criterion specified in GDC-19 (5 rem TEDE in the control room), and meet the dose acceptance criteria specified in the SRP 15.0.1 (6.3 rem TEDE at the EAB).

Even though the NRC staff performed its confirmatory dose calculations, the staff's acceptance is based on the licensee's analyses. The results of the licensee's radiological consequence calculations are provided in Table 1 and the major parameters and assumptions used by the licensee and acceptable to the staff are listed in Table 3. The radiological consequences at the EAB, at the LPZ, and in the control room as calculated by the licensee are also within the dose criterion specified in GDC-19 and meet the dose acceptance criterion specified in the SRP 15.0.1. Therefore, the staff concludes that the proposed AST implementation revising the current design basis radiological consequence analysis for the postulated FHA is acceptable.

3.3 Main Steam Line Break (MSLB) Accident

This DBA postulates a failure in one of the four MSLs at a location outside of containment and downstream of the outermost isolation valve, resulting in the release of steam from both ends of the break. The radiological consequences of a break outside containment bounds those results from a break inside containment. Thus, only the break outside containment is analyzed. The MSIVs are assumed to close in the maximum time allowed by TS (5.5 seconds) and no fuel damage is predicted. The postulated iodine release is based on the maximum primary coolant activity allowed by TS and the noble gas release is based on an off gas release rate of 310 millicuries per second (mCi/s) after 30 minutes decay which conservatively bounds the 290 mCi/s allowed by TS.

The licensee re-evaluated the radiological consequence resulting from the postulated MSLB accident and concluded that the radiological consequences at the EAB, LPZ, and in the control room are within the dose criteria specified in the dose acceptance criteria specified in SRP Section 15.0.1 (6.3 rem TEDE at the EAB) and in 10 CFR 50.67. The NRC staff reviewed the methods, parameters, and assumptions used by the licensee in its radiological dose consequence analyses and finds that they are consistent with the guidance provided in RG 1.183. To verify the licensee's radiological consequence assessments, the staff performed confirmatory radiological consequence dose calculations for the postulated MSLB accident. The radiological consequences calculated by the staff are well within the dose criterion specified in GDC-19 (5 rem TEDE in the control room), and meet the dose acceptance criteria specified in SRP Section 15.0.1. The results of the licensee's radiological consequence calculations are provided in Table 1 and the major parameters and assumptions used by the licensee and acceptable to the staff are listed in Table 4.

3.4 Control Rod Drop Accident (CRDA)

This DBA postulates a high worth control rod drops from its fully inserted or intermediate position in the core. The removal of large negative reactivity from the core results in a localized power excursion. The licensee postulated CRDA at full power will result in a total of 850 fuel rods damaged.

It is assumed that 100 percent of the noble gases but only 10 percent of the iodine released reach the main condenser due to plate out in the reactor pressure vessel and main steam lines. Of the iodine that enters the main condenser, 90 percent plates out. There is no reduction in noble gases. The fission product gases in the main condenser are released at a rate of 1 percent by volume per day as a ground level release via the turbine building. These assumptions are consistent with the guidelines provided in RG 1.183.

The radiological consequences calculated by the licensee and by the NRC staff for the CRDA are within the dose criterion specified in GDC-19 (5 rem TEDE in the control room), and meet the dose acceptance criteria specified in the SRP 15.0.1. The results of the licensee's radiological consequence calculations are provided in Table 1 and the major parameters and assumptions used by the licensee and acceptable to the staff are listed in Table 5.

3.5 Control Room Habitability

The RBS control room is normally maintained at a positive pressure with respect to surrounding air volumes by its normal ventilation system using outside air. The normal outdoor air makeup flow rate is 2000 cfm. The licensee proposed to manually isolate the control room air intakes no later than 20 minutes after the initiation of the postulated LOCA. The normal control room air intake rate of 2000 cfm [without filtration through the control room fresh air (CRFA) system] is assumed for the first 20 minutes. Once the normal air intakes are isolated, the outside air intake will be filtered through the CRFA system and the control room atmosphere is recirculated through the CRFA system at 2000 cfm. The licensee assumed unfiltered air inleakage rate of 300 cfm with filtered intake air flow of 1700 cfm. The CRFA system is a redundant system. Each subsystem consists of, among other things, a pre-filter, a high-efficiency particulate air filter, a charcoal adsorber, and a post-HEPA filter. The NRC staff assumed a removal efficiency of 99 percent for fission products in particulate form for the HEPA filter and 98 percent for iodine in elemental and organic form for charcoal adsorbers (4-inch depth).

The licensee proposed and the NRC staff used in its confirmatory dose calculation an unfiltered air inleakage rate of 300 cfm into the control room during the entire 30-day accident period while the control room is isolated. RBS has not performed an integrated control room unfiltered air inleakage test. However, the staff is currently working toward resolution of generic issues related to control room habitability. The staff's acceptance of the 300 cfm unfiltered air inleakage assumption in this application does not preclude any future generic regulatory actions that may become applicable to RBS.

This assessment may be used in subsequent amendments; however, any use of this assessment that involves a relaxation in requirements will require verification (in accordance with the aforementioned resolution of the generic issues related to control room habitability) that the unfiltered inleakage rate is within the limits of the AST assessment.

The licensee re-evaluated the control room habitability with the application of the AST and concluded that the radiological consequence to the control room operator resulting from the postulated LOCA is within the 5 rem TEDE criterion specified in 10 CFR 50.67. The licensee reached this conclusion:

(1) using the revised atmospheric relative concentrations at the control room air intake (see Section 3.6.3),
(2) with manual isolation of control room at 20 minutes after the initiation of the postulated LOCA,
(3) with an unfiltered air inleakage of up to 300 cfm into the control room during the entire period of the accident while the control room is isolated, and
(4) taking credit for dual manual air intake to the control room (see Section 3.6.3).

To verify the licensee's radiological consequence analyses, the NRC staff performed its confirmatory radiological consequence dose calculation and found its results are also within the dose criteria specified in 10 CFR 50.67 for the control room operator. The results of the licensee's radiological consequence dose calculation are provided in Table 1 and the major parameters and assumptions used by the licensee and the staff are listed in Tables 2 through 9.

Therefore, the NRC staff concludes that adequate radiation protection is provided to the control room operator to permit access to and occupation of the control room under accident conditions without personnel receiving radiation exposures exceeding a TEDE of the 5 rem dose criterion specified in 10 CFR 50.67. Therefore, the staff finds that the control room habitability assessment performed by the licensee is acceptable.

3.6 Atmospheric Relative Concentration (χ/Q values) Estimates

3.6.1 Meteorological Data

The licensee used 5 years of hourly onsite meteorological data collected during calendar years 1995 through 1998 and 2000 to estimate the atmospheric relative concentration values used in the control room dose assessments described above. These data were measured at 9.1 and 45.7 meters above grade at the RBS site. The hourly 1999 data were not used due to a failure of the tapes containing the data. The licensee stated that the program meets the criteria in RG 1.23, "Onsite Meteorological Programs." Other than that, the accuracy of wind speeds greater than 5 miles per hour meets the guidance in RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." The area around the tower has been kept free of obstructions that could affect wind measurements. The instrumentation is maintained in accordance with the plant technical requirements manual that requires a channel check once every 24 hours and calibration every 184 days at which time a walkdown is also typically performed. The measurement system has an uninterrupted power supply and redundancy to enhance data recoverability. The meteorological data are typically checked each working day. Procedural guidance is used to determine acceptability of the data based upon checks for factors such as continuity and/or abnormalities and comparisons with data from redundant channels and previously collected data.

The NRC staff performed a review of the data using the methodology described in NUREG-0917, “Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data,” on meteorological data quality assurance. Further review was performed using a computer spreadsheet. Other than one outage of more than a week’s duration, reported data outages were of relatively short duration. Examination of the data revealed some occurrence of data remaining unchanged for two or more consecutive hours at a higher frequency than expected due to typical meteorological conditions. This occurred most frequently for the lower level wind speeds and made it difficult for staff to confirm that the data recovery rate met the recommendations of RG 1.23.

Regarding atmospheric stability, with one exception, the consecutive-hour occurrence of any single unstable class was of short duration as would be expected by typical meteorological processes. In addition, with few exceptions, stable and neutral conditions were consistently reported to occur at night and unstable and neutral conditions during the day. The 1995 upper level wind data appeared to have a very high occurrence of light winds considering the height of measurement and when compared with data from the other years. In addition, in 1995, the upper level wind speed was slower than the lower level speed much more frequently than expected due to typical meteorological processes and than occurred the other four years. Further, the maximum wind speeds reported at both heights in 1995 were notably lower than during the remaining 4 years. With the exceptions noted, wind speed and direction frequency occurrence at the lower level was reasonably similar from year to year. As the upper level wind data were not directly used in the χ/Q calculations discussed below, any apparent anomalies in those data are not of significance with regard to this amendment.

3.6.2 Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) Relative Concentration Estimates

For the LOCA, CRDA, and MSLB accident, the licensee calculated χ/Q values for the EAB and LPZ using site-specific inputs and the PAVAN computer code. The PAVAN code, documented in NUREG/CR-2858, “PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Plants,” uses the methodology described in RG 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.” χ/Q values were calculated for postulated releases from the Main Plant Stack (Standby Gas Treatment System), the Main Steam Blowout Panel and the Turbine Building. Distances used in the analyses were based upon the shortest distance from each postulated release point to the EAB and LPZ. All releases were assumed to be ground level. The licensee used an average joint frequency distribution based upon data from the period 1994 through 1999. Since the data tended to be somewhat clustered in several wind speed categories, the NRC staff performed comparative estimates using the 1995 through 1998 and 2000 data discussed above. Resultant χ/Q estimates were not significantly different than those calculated by the licensee.

For the FHA dose assessment, the licensee used licensing basis values based upon RG 1.145 and 2 years of meteorological data measured in the late 1970's. The licensee performed comparative calculations to demonstrate that use of the licensing basis χ/Q values was more limiting than using those calculated based upon the PAVAN computer code and data from 1994 through 1999.

3.6.3 Control Room Relative Concentration Estimates

For the LOCA, CRDA and MSLB accident, the licensee used 1995 through 1998 and 2000 meteorological data and the ARCON96 methodology (NUREG/CR-6331, Revision 1, “Atmospheric Relative Concentrations in Building Wake”) to calculate χ/Q values for control room dose assessment with a modification to the surface roughness length and averaging sector width constant. Both modifications are acceptable to the NRC staff. All postulated releases were considered as ground level point releases assuming no vent flow and thus utilized the lower level wind data in the calculations.

The RBS facility is designed with dual control room air intakes and the capability to manually select between intakes. As part of this license amendment, the licensee has modified the licensing basis methodology used to calculate the LOCA, CRDA and MSLB accident control room χ/Q values. The previous licensing basis procedure was based upon the methodology described in SRP 6.4, “Control Room Habitability System.” Using this methodology, the χ/Q value at the more favorable intake location was reduced by a factor of four to account for dilution effects associated with a dual intake configuration and the relative probability that the operator would make the proper intake selection.

As a part of this amendment, the licensee modified this procedure to calculate effective χ/Q values weighted by assumed inflow rates to account for possible undesired inleakage into the control room. To do so, the licensee assumed a design inflow rate of 1700 cfm from the favorable intake and an undesired control room inleakage rate of 300 cfm from the main intake, resulting in a total inflow rate of 2000 cfm. Each effective χ/Q value was then calculated by first multiplying the favorable χ/Q value that had been reduced by a factor of 4 by 1700 cfm. This value was then added to the main air intake χ/Q value multiplied by 300 cfm and the resultant value was divided by 2000 cfm to result in the flow weighted effective χ/Q value. The NRC staff notes that it may be necessary to recalculate the effective χ/Q values if the assumed flow rates change.

In the FHA dose assessment, the licensee used licensing basis χ/Q values calculated based upon the Murphy-Campe methodology referenced in SRP 6.4. The licensee also made comparative estimates to demonstrate that the χ/Q values calculated using the Murphy-Campe methodology were more limiting than χ/Q values calculated using the revised effective χ/Q methodology described above in the previous paragraph.

The NRC staff qualitatively reviewed the inputs to the ARCON96 calculations and found them to be consistent with site configuration drawings and staff practice. The staff also made several confirmatory estimates of the licensee’s calculations. In addition, the staff attempted to make comparative calculations to assess the impact of the use of the 1995 data given the uncertainties discussed above and the one year gap in the meteorological data between 1998 and 2000. As a result, the staff has judged that the χ/Q values calculated by the licensee are adequate for use in the dose assessment described above.

4.0 PROPOSED TECHNICAL SPECIFICATIONS CHANGES

4.1 TS Section 1.1, "Definitions"

The licensee proposed to re-define "Dose Equivalent I-131" to state, "The thyroid dose conversion factors used for this calculation shall be those listed in Federal Guidance Report No. 11, Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," deleting reference to TID-14844. This proposal is consistent with the NRC staff technical position stated in Regulatory Issue Summary (RIS) 01-019, "Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests," and Section 1.1, "Definition of NUREG-1434, Revision 2, Standard Technical Specifications - General Electric Plants."

In NRC RIS 01-019, the NRC staff endorsed the use of iodine dose conversion factors listed in the International Commission on Radiological Protection Publication No. 30 (ICRP-30), "Limits for Intakes of Radionuclides by Workers, 1979," in determining the iodine-131 dose equivalent reactor coolant activity in TS and in calculating the radiological consequences DBAs. The iodine dose conversion factors in ICRP-30 are the same as those tabulated in Federal Guidance Report No. 11. Therefore, the staff finds that this proposed change is acceptable.

4.2 TS Section: 3.3.7.1, Control Room Fresh Air (CRFA) System Instrumentation

The licensee proposed to amend TS Section 3.3.7.1, to revise Note (b) in Table 3.3.7.1-1 to reflect that the requirements are only applicable during movement of "recently irradiated fuel." The licensee provided the following justification:

> Filtration by the CRFA system is currently credited in all of the Design Basis Accidents: LOCA, FHA, Control Rod Drop Accident (CRDA), and MSLB. These analyses credit initiation of the system via either a LOCA signal (reactor water level 2, high drywell pressure, etc.) or via a high radiation signal from the intake radiation monitors. These signals tied with automatic initiation of the CRFA filters are the basis for this TS. The Bases for this section states "The ability of the CRFA system to maintain the habitability of the MCR [Main Control Room] is explicitly assumed for certain accidents as discussed in the USAR safety analyses."
>
> The FHA (including the Light Load Drop Accident (LLA) analysis) and MSLB dose analyses do not credit filtration by the CRFA charcoal filters. However, those analyses assume a minimum decay time of 24 hours prior to an FHA potentially occurring. Currently "recently irradiated fuel" is defined in the TS bases (Section 3.6.4.5) as "fuel which has been part of a critical reactor core within the previous 11 days." The FHA analysis prepared in support of AST effectively redefines "recently irradiated fuel" based on a 24 hour decay time. Therefore, in the unlikely event fuel may be moved prior to 24 hours this specification would directly apply and the MCR Local Air Intake radiation monitors would be required to ensure that the potential dose to MCR operators meets 10 CFR 50.67 dose limits. It should be noted that a decay time of 24 hours is currently required prior to fuel movement per Technical Requirements Manual (TRM) Section 3.9.10.

> One CRDA scenario credited the CRFA system; however, it assumed manual initiation of the system 20 minutes into the event. Both air intakes for the main control room have redundant radiation monitors which annunciate in the main control room. Automatic initiation of the system is not assumed. The calculated dose consequences of a CRDA meet the acceptance criteria from 10 CFR 50.67 and RG 1.183. The CRFA filters are also credited in the LOCA dose analyses (Attachment 7). That analysis also conservatively assumes manual initiation of the system, however, no relaxation of the automatic initiation requirements of the system is requested at this time. All doses meet the acceptance criteria of 10 CFR 50.67 and RG 1.183.

The NRC staff has reviewed the licensee's proposed change in the applicability requirements for CRFA system instrumentation. Based on the staff's acceptance of the licensee's proposed AST implementation and revised design basis radiological consequence analyses and the applicability of these requirements only during movement of recently irradiated fuel, the staff finds this proposed TS change acceptable.

4.3 TS Section 3.6.1.2, "Primary Containment Air Locks"

The licensee proposed to revise: (1) SR 3.6.1.2.1 to remove requirements for annulus bypass leakage, and (2) SR 3.6.1.2.4 to increase the leakage rate for the air lock seal pneumatic system from 1.28 psig (pounds per square inch gauge)/day to 1.5 psig/day. The licensee provided the following justification:

> Annulus bypass leakage is leakage which will bypass the annulus and be released into the Auxiliary Building. All leakage paths (including the containment/fuel building personnel air lock and Inclined Fuel Transfer System [IFTS] drain line) which can potentially bypass secondary containment are currently included in the Secondary Containment Bypass (SCB) summation (See TRM Table 3.6.1.1-1 for a list of Annulus Bypass penetrations). Since the "annulus bypass" leakage paths now lead to the Auxiliary Building, the air would be filtered by SGTS prior to release to the environment. The AST LOCA dose analysis does not assume an explicit "annulus bypass" leakage path. RBS performed a sensitivity study demonstrating that off-site doses are not sensitive to annulus bypass leakage as long as such leakage paths are filtered. Annulus bypass leakage is only a small po[r]tion of the overall containment leakage term and both the Auxiliary Building and the Annulus are filtered by SGTS. Therefore, the impact to calculated AST doses is negligible and simplifying the model is acceptable. Thus, the annulus bypass leakage summation is no longer required and should be deleted from Technical Specifications.
>
> The primary containment personnel air locks (PAL) are used to ingress and egress the primary containment. The PAL doors have a seal which remains pressurized via the seal air system. Technical Specification SR 3.6.1.2.2 states that the minimum normal operation pressure for the primary containment air lock seal air flask pressure must be $\geq$ 90 psig. The inflatable seals must remain above a pressure of 45 psig to maintain its integrity. The current allowable leakage rate of 1.28 psi [pounds per square inch] per 24 hours is based on a period of 35 days ((90 psig - ((90 psig - 45 psig) / 35 days = 1.286 psi/day). Since dose calculations are

> performed for a period of 30 days a value of 1.5 psi/day is requested ((90-45)/30=1.5).

The NRC staff accepts the licensee's justification that the AST LOCA radiological consequences analysis is not sensitive to annulus bypass leakage as long as such leakage paths are filtered, which they are, in this case, by the SGTS. Therefore, the staff agrees that the annulus bypass leakage summation is no longer required and may be deleted from TSs. However, as the licensee stated, annulus bypass leakage is a part of the overall containment leakage term. As such, this TS change does not change the containment leakage rate testing requirements of Appendix J to 10 CFR Part 50 or TS 5.5.13, "Primary Containment Leakage Rate Testing Program," as they relate to the annulus bypass penetrations.

Regarding the PAL door seal allowable leakage rate of SR 3.6.1.2.4, the isolation capability of the PAL is not required to be operable for more than 30 days. Thus, the NRC staff finds the recalculated leakage rate limit to be appropriate and acceptable.

4.4 <u>TS Section 3.6.1.3, "Primary Containment Isolation Valves"</u>

The licensee proposed to amend SR 3.6.1.3 to: (1) revise SR 3.6.1.3.9 to increase the allowable secondary containment bypass leakage rate from "$\leq$170,000 cc/hr when pressurized to $\geq P_a$" to "$\leq$580,000 cc/hr when pressurized to $\geq P_a$," (2) revise SR 3.6.1.3.10 to set a single MSL leakage limit of 50 scfh when tested at $\geq P_a$, and (3) delete SR 3.6.1.3.12 (annulus bypass leakage rate summation).

The licensee provided the following justification:

> Secondary containment bypass leakage is leakage from the primary containment building which will bypass the annulus and auxiliary building, thus it will potentially escape to the environment unfiltered. This leakage is independent of the overall containment rate summation (L_a). Prior to RBS TS Amendment 98 the Penetration Valve Leakage Control System (PVLCS) system was assumed to be manually initiated which terminated this release path early in the event. The PVLCS system was deleted via TS Amendment 98. The Inclined Fuel Transfer System (IFTS) drain line and the fuel building personnel air lock were added since the fuel building was removed from the secondary containment envelope via RBS TS Amendment 113. A review of the potential leakage paths for this summation concluded that a value of 580,000 cc/hr was appropriate for use in the AST LOCA Analysis (Attachment 7). The analysis dose consequences using this value met the criteria set forth in 10 CFR 50.67.
>
> Annulus bypass leakage is leakage which will bypass the annulus and be released into the Auxiliary Building. All leakage paths (including the containment/fuel building personnel air lock and IFTS drain line) which can potentially bypass secondary containment are currently included in the Secondary Containment Bypass (SCB) summation (TS SR 3.6.1.3.9). Since the "annulus bypass" leakage paths now lead to the Auxiliary Building, the air would be filtered by SGTS prior to release to the environment. The AST LOCA dose analysis does not assume an explicit "annulus bypass" leakage path. RBS performed a sensitivity study demonstrating that off-site doses are not sensitive to annulus bypass leakage as long as such leakage paths

> are filtered. Annulus bypass leakage is only a small portion of the overall containment leakage term, and both the Auxiliary Building and the Annulus are filtered by SGTS. Therefore, the impact to calculated AST doses is negligible and simplifying the model is acceptable. Thus, the annulus bypass leakage summation is no longer required and should be deleted from Technical Specifications.
>
> The current TID [total integrated dose] analysis does not assume a failure of an MSIV. The AST analysis conservatively assumed that one MSIV failed (in addition to an EDG [emergency diesel generator]). The leakage rate of 50 scfh was assumed through the steam line containing the failed MSIV. This value corresponds to the proposed limit to be incorporated in SR 3.6.1.3.10.

The first two changes, for the allowable secondary containment bypass leakage rate and the single main steam line leakage rate limit, are consistent with the AST radiological consequences analysis. There are no other safety considerations involved; the proposed numbers are assumptions used in the AST radiological consequences analysis, and, insofar as the NRC staff finds the AST radiological consequences analysis to be acceptable, then these two changes are acceptable.

The NRC staff's conclusion regarding the third change, deletion of the annulus bypass leakage rate summation, is the same as presented above in Section 4.2 regarding the evaluation of TS Section 3.6.1.2.

4.5 TS Section: 3.6.4.1, "Secondary Containment - Operating"

The licensee proposed to amend SR 3.6.4.1.4 by revising the Auxiliary Building drawdown time from 13.5 seconds to 34.5 seconds.

The licensee provided the following justification:

> LOCA dose analyses do not credit secondary containment until an adequate vacuum is reached (≤-0.25 in. w.g.). The AST LOCA dose analysis (Attachment 7) conservatively assumed manual initiation of SGTS, even though the automatic initiation function and associated TS requirements are retained. The analysis also allowed an additional 10 minutes for an adequate vacuum to be established. This resulted in a total assumed Positive Pressure Period (PPP) of 30 minutes (20 minutes for Operator action and an additional 10 minutes to establish the required vacuum). Calculations used the GOTHIC computer code to demonstrate that the PPP assumed in the AST LOCA dose analysis are conservative for the annulus and auxiliary building, respectively. They determined a revised drawdown time for testing during non-accident conditions using current system parameters (flow rates, etc.). That analysis determined that an auxiliary building drawdown time of 38.5 seconds will ensure that the calculated PPP is conservative. This submittal requests a value of 34.5 seconds (~ 90% of the calculated value) for conservatism.

In its review, the NRC staff found that the proposed number is consistent with the assumption used in the AST radiological consequences analysis. Insofar as the staff finds the AST radiological consequences analysis to be acceptable, then this change is acceptable.

4.6 TS Section: 3.6.5.1, “Drywell” and 3.6.5.2, “Drywell Air Locks”

The licensee proposed to amend: (1) SR 3.6.5.1.2 to increase the leakage rate for the air lock seal pneumatic system from 0.67 psig/day to 20.0 psig/day, and (2) SR 3.6.5.2.5 to increase the leakage rate for the air lock seal pneumatic system from 0.67 psig/day to 20.0 psig/day.

The licensee provided the following justification:

> Drywell integrity is credited in the containment and drywell pressure response analyses developed in support of the power uprate project approved via TS Amendment 114. The current leakage rate is based on a 30 day duration, i.e., drywell seal integrity is required for at least 30 days following a postulated LOCA. The proposed leakage rate for the drywell air lock seals is based on leakage over the first 24 hours. After 24 hours, the seals could potentially fail due to the internal pressure not being adequate. A large break in the drywell would increase pressure rapidly and uncover the suppression pool vents. Containment pressure would increase steadily as a result of flow from the drywell. Following this initial blowdown period, the pressure in the drywell begins to drop due to steam condensation. Eventually the drywell pressure drops below the containment pressure and becomes a relative vacuum (see figures in RBS USAR Section 6.2). This occurs well before 24 hours. Failure of the seals would tend to reduce containment pressure to less than what is analyzed since the drywell is at a lower pressure. Thus, the analyses for large breaks are conservative since the pressures they calculate bound the pressures that would be expected if the drywell air lock seals failed.
>
> Intermediate and Small Break Accidents were also evaluated. The "short term" analyses show the drywell pressure is greater than containment pressure, however, the "long term" (>10,000 seconds) analyses show the drywell pressure drops below containment pressure well before 24 hours. The reason for the differences lies with computer code limitations and decay heat assumptions. Since the drywell has a lower pressure than the containment 24 hours into the event, the calculated containment pressures are conservative and bound seal failure at 24 hours.
>
> The AST LOCA dose analysis (Reference Attachment 7) treats the drywell and containment as two separate nodes. Early in the event (<20 minutes) the flow rate between the drywell and containment is based on the containment pressure response to a recirculation line break. RG 1.183, Appendix A, Section 3.7 states, "After two hours, the radioactivity is assumed to be uniformly distributed throughout the drywell and the primary containment." Since the containment and drywell are homogenized at 2 hours as required by Regulatory Guide 1.183, and seal failure would not occur until after 24 hours, there is no impact to the LOCA doses calculated using AST based assumptions.
>
> The air lock seals remain pressurized via the seal air system. Technical Specification SR 3.6.5.1.1 and SR 3.6.5.2.2 require that the minimum normal operating pressure for drywell seal air flask pressure must be ≥ 75 psig. The inflatable seals must remain above a pressure of 55 psig to maintain its integrity. The current allowable leakage rate of 0.67 psi per 24 hours is based on a period of 30 days ((75psig - 55psig)/30days = 0.67 psi/day). The discussion above

> demonstrates that the seals are not required after 24 hours, therefore, a value of 20.0 psi per 24 hours is requested ((75psi - 55psi)/1 day= 20.0 psi/day).

The NRC staff agrees that, per the safety analysis, drywell leakage integrity is not needed after the first 24 hours of an accident. Therefore, the staff finds the proposed changes, which would assure drywell air lock door seal operability out to 24 hours, to be acceptable.

4.7 TS Section: 3.7.2, “Control Room Fresh Air (CRFA) System”

The licensee also proposed to amend TS Section 3.7.2, to revise it to: (1) reflect that the requirements are only applicable during movement of “recently irradiated fuel,” (2) reflect that CONDITION C and its associated REQUIRED ACTIONS are consistent with the APPLICABILITY change concerning movement of “recently irradiated fuel,” and (3) reflect that CONDITION E and its associated REQUIRED ACTIONS are consistent with the APPLICABILITY change concerning movement of “recently irradiated fuel.” The licensee provided the following justification:

> Filtration by the CRFA system is currently credited in all of the Design Basis Accidents: LOCA, FHA, CRDA, and MSLB. These analyses credit initiation of the system via either a LOCA signal (reactor water level 2, high drywell pressure, etc.) or via a high radiation signal from the intake radiation monitors. These signals tied with automatic initiation of the CRFA filters is the basis for this TS.
>
> The MSLB dose analyses do not credit filtration by the CRFA charcoal filters. The FHA analysis also does not credit the CRFA charcoal filters. However, it inherently assumes a decay time of 24 hours prior to an FHA potentially occurring. Therefore, during movement of “recently irradiated fuel” the TS and applicable SR will be applicable since operation of the charcoal filter train would be required to ensure applicable doses are bounded. As such the definition of “recently irradiated fuel” will be tied to the 24 hour decay time assumed in the AST FHA analysis. It should be noted that movement of irradiated fuel with decay times <24 hours is currently prohibited at RBS.
>
> The CRFA filters are credited in the LOCA dose analysis (Reference Attachment 7). That analysis assumes manual initiation of the system. Review of system design indicates that in actuality the CRFA system will be operating in emergency mode prior to the onset of fuel damage. Specifically, the current TID LOCA analysis assumes that the system starts automatically and credits ESF charcoal filtration 66 seconds into the event, whereas the AST dose methodology does not assume fuel damage until 2 minutes into the event. Therefore, assuming manual initiation of the system at 20 minutes is clearly conservative.
>
> The CRDA analysis (Reference Attachment 9) also credited the CRFA system, however, it conservatively assumed manual initiation rather than automatic initiation of the charcoal filters. Fuel damage for a CRDA is assumed to occur at the onset of the event. Once again, however, the 20 minutes assumed for manual initiation of the system easily bounds the initiation time assumed in the current CRDA analysis of 66 seconds.

The NRC staff has reviewed the licensee’s proposed change in the applicability requirements for the CRFA system. Based on the staff’s acceptance of the licensee’s proposed AST implementation and revised design basis radiological consequence analyses and the applicability of these requirements only during movement of recently irradiated fuel, the staff finds this proposed TS change acceptable.

4.8 TS Section: 3.7.3, “Control Room AC System”

The licensee proposed to amend TS Section 3.7.3, “Control Room AC System,” to revise it to: (1) reflect that the requirements are only applicable during movement of “recently irradiated fuel,” (2) reflect that CONDITION D and its associated REQUIRED ACTIONS are to be consistent with the APPLICABILITY change concerning movement of “recently irradiated fuel,” and (3) reflect that CONDITION E and its associated REQUIRED ACTIONS are to be consistent with the APPLICABILITY change concerning movement of “recently irradiated fuel." The licensee provided the following justification:

> Technical Specifications Bases for Section 3.7 explains that the intent of the section is to ensure that required equipment is not affected by adverse environmental conditions, specifically, high temperatures. The applicability discussion states that, “In Modes 4 and 5, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations of these modes.[”] Therefore, maintaining the Control Room AC System OPERABLE is not required in Modes 4 and 5 except when a significant radiological release is available. The FHA no longer credits mitigation by any system including the CR AC and CRFA Systems. However, a decay time of 24 hours is assumed, therefore, the TS would remain applicable during movement of “recently irradiated fuel.” The dose consequences meet the criteria set forth in 10 CFR 50.67 and Regulatory Guide 1.183. Therefore, this TS is no longer applicable.

The NRC staff has reviewed the licensee's proposed change in the applicability requirements for Control Room AC System. Based on the staff's acceptance of the licensee's proposed AST implementation and revised design basis radiological consequence analyses and the applicability of these requirements only during movement of recently irradiated fuel, the staff finds this proposed TS change acceptable.

4.9 TS Section: 5.5.7, “Ventilation Filter Testing Program (VFTP)”

In its application dated April 24, 2002, the licensee proposed to amend TS Section 5.5.7 to: (1) delete the fuel building ventilation system (FBVS) requirements, (2) revise the SGTS allowable penetration from 0.5 percent to 5.0 percent, and (3) revise the CRFA allowable penetration from 0.5 percent to 1.0 percent. The licensee provided the following justification for this portion of its request:

> RBS committed to ASTM D3803-1989 via Technical Specification Amendment 115 which allows licensees to test to 50% of the margin assumed in the safety analyses. SGTS is only credited in the LOCA doses analysis which assumes 90% filter efficiency, therefore, the testing acceptance criteria is being revised to 5%. The CRFA filters are credited in the LOCA and the CRDA analyses. These analyses assume a filter efficiency of 98% which mandates a testing acceptance criteria of

> 1%. **The changes proposed with respect to the Fuel Building ventilation system in reference 1 are withdrawn.** (Bold type added by NRC staff to indicate a change between the licensee's submittals dated April 24, 2002, and December 18, 2002.)

The NRC staff reviewed the proposed changes to TS 5.5.7, noting that the licensee withdrew the proposed change that deletes the FBVS requirements. The staff position, as defined in RG 1.52, Revision 3, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," is that a safety factor of two is acceptable for the filter testing acceptance criteria. The proposed changes for the SGTS filter acceptance criteria and the CRFA filter acceptance criteria both are in compliance with this guidance. The staff concludes that the proposed test criteria will provide reasonable assurance that the performance of the filters will be adequate to protect the health and safety of the public and control room operators.

4.10 TS 5.5.13, "Primary Containment Leakage Rate Testing program"

The licensee proposed to amend TS Section 5.5.13 to increase the containment leakage rate from 0.26 percent per day to 0.325 percent per day.

The licensee provided the following justification:

> The containment leakage rate is a major assumption in the LOCA dose analysis. The AST LOCA analysis (Reference Attachment 7) assumed a containment leakage rate of 0.325% per day. The dose consequences of that analysis met the criteria set forth in 10 CFR 50.67, and, therefore, the increased leakage rate is acceptable.

The NRC staff reviewed the proposed revision and determined that this change is consistent with the AST radiological consequences analysis. There are no other safety considerations involved, although SRP Section 6.2.6, "Containment Leakage Testing," states that L_a, the allowable containment leakage rate, should not be less than 0.1 percent per day (because of testing accuracy), which the proposed number satisfies. The proposed number is an assumption used in the AST radiological consequences analysis, and, insofar as the staff finds the AST radiological consequences analysis to be acceptable, this change is acceptable.

4.11 TS Section 3.8.2, "AC Sources - Shutdown," TS Section 3.8.5, "DC Sources - Shutdown," TS Section 3.8.8, "Inverters - Shutdown," and TS Section 3.8.10, "Distribution Systems - Shutdown"

In TS sections 3.8.2, 3.8.5, 3.8.8, and 3.8.10, the licensee proposed to revise the APPLICABILITY statement from "during movement of irradiated fuel," to "during movement of recently irradiated fuel," revise the REQUIRED ACTIONS Condition A to be consistent with the APPLICABILITY change concerning movement of "recently irradiated fuel," and revise the REQUIRED ACTIONS associated with Condition B to be consistent with the APPLICABILITY change concerning movement of "recently irradiated fuel."

The licensee provided the following justification for each of these proposed changes:

> The TS Bases for this section explains that this TS is applicable to ensure that "Systems needed to mitigate a fuel handling accident are available." Previously, the CRFA and fuel building ventilation systems' charcoal filter trains were credited to mitigate the consequences of a FHA. The AST analyses do not credit these or any other systems to mitigate consequences of a FHA. Based on the above discussion it can be concluded that accident assumptions will be met, therefore, this TS is no longer applicable except during movement of "recently irradiated fuel."

Based on the considerations discussed above, the NRC staff has reviewed the licensee's submittal and concluded that there is reasonable assurance that accident assumptions will be met, and these TSs are no longer applicable except during the movement of recently irradiated fuel. Therefore, these portions of the TSs may be revised as proposed by the licensee. Also, this is in compliance with the 10 CFR 50.67 and RG 1.183 and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

4.12 Technical Conclusion

The NRC staff has reviewed the licensee's analyses and performed confirmatory assessments of the radiological consequence of the postulated four DBAs. The doses calculated by the licensee are listed in Table 1. The doses calculated by the licensee are all within the dose criteria set forth in 10 CFR 50.67 and the dose acceptance criteria specified in Table 1 of SRP 15.0.1. The staff has independently confirmed the licensee's dose calculation. Therefore, the staff concludes that the radiological consequences analyzed and submitted by the licensee are acceptable.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (67 FR 40021 dated June 11, 2002). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Attachments:
1. Table 1
2. Table 2
3. Table 3
4. Table 4
5. Table 5
6. Table 6
7. Table 7
8. Table 8
9. Table 9

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Table 1
Radiological Consequences Expressed as TEDE
(rem)

Design Basis Accidents	EAB[1]	LPZ[2]	Control Room
LOCA			
Containment leakage pathway	2.60	1.70	0.40
ECCS leakage pathway	0.34	0.57	0.15
MSIV leakage pathway	12.30	5.40	2.90
Total	15.24	7.67	3.45
Dose acceptance criteria[3]	25	25	5.0
Fuel handing accident	2.44	0.32	1.62
Dose acceptance criteria	6.3[4]	6.3[4]	5.0[3]
Main steam line break			
Pre-accident spike	1.4	0.2	2.2
Dose acceptance criteria[3]	25	25	5.0
Equilibrium state	<0.1	<0.1	0.2
Dose acceptance criteria	2.5[4]	2.5[4]	5.0[3]
Control rod ejection accident	0.9	0.4	4.3
Dose acceptance criteria	6.3[4]	6.3[4]	5.0[3]

(1) Exclusion area boundary
(2) Low population zone
(3) 10 CFR 50.67
(4) SRP 15.0.1

Table 2
Parameters and Assumptions Used in
Radiological Consequence Calculations
Loss-of-Coolant Accident

Parameter	Value
Reactor power	3100 MWt
Building volumes, ft^3	
Containment	1.19E+6
Drywell	2.36E+5
Annulus	3.57E+5
Auxiliary	1.16E+6
Suppression pool	1.25E+5
Containment leak rates	
0 to 1 hour	0.325% per day
1 to 720 hours	0.179% per day
Secondary containment bypass leak rates	
0 to 1 hour	0.341 cfm
1 to 720 hours	0.188 cfm
Main steam line leak rates	
0 to 25 minutes	50 cfh
25 minutes to 30 days	0
Secondary containment positive pressure period	30 minutes
Standby gas treatment system filter efficiencies	
Particulate	99%
Iodine	90%
Standby gas treatment system flow rates	
Annulus building release	2.5E+3 cfm
Auxiliary building release	1.0E+4 cfm
ECCS leak rate	
0 to 720 hours	1 gallon per minute
Iodine partition factor	10%
Control room	
Volume 1.88E+5 ft^3	
Filtered makeup air flow	1.7E+3 cfm
Filtered Recirculation air flow	2000 cfm
Unfiltered air inleakage rate	300 cfm
Filter efficiencies	
Aerosol	99%
Elemental iodine	98%
Organic iodine	98%

Table 3
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Fuel Handling Accident

Parameter	Value
Reactor power	3100 MWt
Radial peaking factor	2.0
Fission product decay period	24 hours
Number of fuel rod damaged	150
Fuel pool water depth	23 ft
Fuel gap fission product inventory	
Noble gases excluding Kr-85	5%
Kr-85	10%
I-131	8%
Alkali metals	12%
Fuel pool decontamination factors	
Iodine	200
Noble gases	1
Duration of accident	2 hours
Fission product release point	Fuel building as ground level release
Control room	
Volume	1.88E+5 ft^3
Filtered makeup air flow	1.7E+3 cfm
Filtered Recirculation air flow	2000 cfm
Unfiltered air inleakage rate	300 cfm
Filter efficiencies	
Aerosol	Not credited
Elemental iodine	Not credited
Organic iodine	Not credited

Table 4
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Main Steam Line Break Accident

Parameter	Value
Reactor power	3100 MWt
Reactor coolant activity	
Equilibrium iodine concentration	0.2 µCi/gm dose equivalent I-131
Pre-accident iodine spike case	4 µCi/gm dose equivalent I-131
Noble gas activity	3.1E+5 µCi/second at 30 minutes
Break isolation time	5.5 seconds
Mass releases	
Steam	1.16E+4 lbm
Liquid	6.89E+4 lbm
Fission product release rate	Instantaneous as ground level release
Release point	Main steam tunnel blowout panel
Control room	
Volume	1.88E+5 ft^3
Filtered makeup air flow	1.7E+3 cfm
Filtered Recirculation air flow	2000 cfm
Unfiltered air inleakage rate	300 cfm
Filter efficiencies	
Aerosol	Not credited
Elemental iodine	Not credited
Organic iodine	Not credited

Table 5
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Control Rod Drop Accident

Parameter	Value
Reactor power	3100 MWt
Number of fuel rods damaged	850
Fuel gap activity fractions	
Nobel Gases	10%
Iodine	10%
Fuel melt fraction	
Noble gases	100%
Iodine	50%
Condenser volume	1.06E+5 ft^3
Release fractions from condenser	
Noble gases	100%
Iodine	10%
Particulate	1%
Release rate from condenser	1% per day
Release duration from condenser	24 hours
Control room	
Volume	1.88E+5 ft^3
Filtered makeup air flow	1.7E+3 cfm
Filtered Recirculation air flow	2000 cfm
Unfiltered air inleakage rate	300 cfm
Filter efficiencies	
Aerosol	Not credited
Elemental iodine	Not credited
Organic iodine	Not credited

Table 6
Meteorological Data Used for
LOCA Dose Assessments

Exclusion Area Boundary

Time (hr)	Release Location	χ/Q (sec/m^3)
0 - 2	SGTS*	6.05 E-4**
0 - 2	Turbine Building	7.51 E-4

Low Population Zone Distance

Time (hr)	Release Location	χ/Q (sec/m^3)
0 - 8	SGTS	7.49 E-5
8 - 24	SGTS	5.02 E-5
24 - 96	SGTS	2.10 E-5
96 - 720	SGTS	6.13 E-6
0 - 8	Turbine Building	7.79 E-5
8 - 24	Turbine Building	5.23 E-5
24 - 96	Turbine Building	2.21 E-5
96 - 720	Turbine Building	6.40 E-6

Control Room

Time (hr)	Release Locations	X/Q (sec/m^3)
0 - 2	SGTS	2.53 E-4
2 - 8	SGTS	1.88 E-4
8 - 24	SGTS	8.06 E-5
24 - 96	SGTS	6.21 E-5
96 - 720	SGTS	5.10 E-5
0 - 2	Turbine Building	5.31 E-4
2 - 8	Turbine Building	4.36 E-4
8 - 24	Turbine Building	1.85 E-4
24 - 96	Turbine Building	1.58 E-4
96 - 720	Turbine Building	1.16 E-4

* SGTS - standby gas treatment system
** 6.05 E-4 represents 6.05 x 10^{-4}

Table 7
Meteorological Data Used for
FHA Dose Assessments

Exclusion Area Boundary

Time (hr)	Release Location	X/Q (sec/m^3)
0 - 2	Containment	8.58 E-4

Low Population Zone Distance

Time (hr)	Release Location	X/Q (sec/m^3)
0 - 8	Containment	1.13 E-4

Control Room

Time	Release Location	X/Q (sec/m^3)
0 - 20 min	Containment	1.62 E-3
20 min - 8 hr	Containment	4.05 E-4
8 - 24	Containment	3.00 E-4
24 - 96	Containment	1.01 E-4
96 - 720	Containment	1.62 E-5

Table 8
Meteorological Data Used for
CRDA Dose Assessments

Exclusion Area Boundary

Time (hr)	Release Location	X/Q (sec/m^3)
0 - 2	Turbine Building	7.51 E-4

Low Population Zone Distance

Time (hr)	Release Location	X/Q (sec/m^3)
0 - 8	Turbine Building	7.79 E-5
8 - 24	Turbine Building	5.23 E-5
24 - 96	Turbine Building	2.21 E-5
96 - 720	Turbine Building	6.40 E-6

Control Room

Time (hr)	Release Location	X/Q (sec/m^3)
0 - 2	Turbine Building	5.31 E-4
2 - 8	Turbine Building	4.63 E-4
8 - 24	Turbine Building	1.85 E-4
24 - 96	Turbine Building	1.58 E-4
96 - 720	Turbine Building	1.16 E-4

Table 9
Meteorological Data Used for
MSLB Accident Dose Assessments

Exclusion Area Boundary

Time (hr)	Release Location	X/Q (sec/m^3)
0 - 2	MST* Blowout Panel	6.33 E-4

Low Population Zone Distance

Time (hr)	Release Location	X/Q (sec/m^3)
0 - 8	MST Blowout Panel	7.57 E-5
8 - 24	MST Blowout Panel	5.08 E-5
24 - 96	MST Blowout Panel	2.13 E-5
96 - 720	MST Blowout Panel	6.24 E-6

Control Room

Time (hr)	Release Location	X/Q (sec/m^3)
0 - 8	MST Blowout Panel	3.64 E-3
8 - 24	MST Blowout Panel	2.69 E-3
24 - 96	MST Blowout Panel	1.46 E-3
96 - 720	MST Blowout Panel	2.73 E-4

* MST - main steam tunnel

River Bend Station

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