September 24, 2004

Mr. J. W. Moyer, Vice President
Carolina Power & Light Company
H. B. Robinson Steam Electric Plant
Unit No. 2
3581 West Entrance Road
Hartsville, South Carolina 29550

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 - ISSUANCE OF AN AMENDMENT ON FULL IMPLEMENTATION OF THE ALTERNATIVE SOURCE TERM (TAC NO. MB5105)

Dear Mr. Moyer:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 201 to Renewed Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2). This amendment changes the HBRSEP2 Technical Specifications (TS) in response to your request dated May 10, 2002, as supplemented by letters dated March 12, 2003, April 10, 2003, March 5, 2004, and July 22, 2004.

The amendment approves full implementation of the alternative source term (AST) at HBRSEP2 with the exception of the loss-of-coolant accident (LOCA). Appendix A of Regulatory Guide 1.183 describes the manner in which containment spray removal rate constants are to be selected. The manner chosen by you was inconsistent with this guidance. Since you did not provide a sufficient amount of information to allow the NRC staff to reach a conclusion on the acceptability of your proposed method, the use of AST for the LOCA cannot be approved at this time. The enclosed Notice of Partial Denial of Amendment and Opportunity for Hearing has been forwarded to the Office of the Federal Register for publication.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

/RA/

Chandu P. Patel, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-261

Enclosures:
1. Amendment No. 201 to DPR-23
2. Safety Evaluation
3. Notice of Partial Denial

cc w/encls: See next page

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cc w/encls: See next page

Package No.: ML042680116
Tech Specs: ML042710206 Nrr-100
Federal Register Notice: ML042680118 AMD-012
ADAMS Accession No.: ML042680089 Nrr-058 *SE dated September 16, 2004

OFFICE	PDII-2/PM	PDII-2/LA	SPSB:DSSA*	PDII-2/SC	OGC
NAME	CPatel	EDunnington	Hayes/Brown	MMarshall	
DATE	9/17/2004	9/17/2004	9/16/2004	09/22/2004	9/22/2004

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AMENDMENT NO. 201 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-23 - H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

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CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-261

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 201
Renewed License No. DPR-23

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

   A. The application for amendment by Carolina Power & Light Company (the licensee), dated May 10, 2002, as supplemented by letters dated March 12, 2003, April 10, 2003, March 5, 2004, and July 22, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 3.B. of Renewed Facility Operating License No. DPR-23 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 201, are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 24, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 201

RENEWED FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

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 Pages	Insert Pages
1.1-2	1.1-2
3.4-35	3.4-35
3.4-45	3.4-45
3.4-46	3.4-46
3.4-48	3.4-48

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 201 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-23

CAROLINA POWER & LIGHT COMPANY

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-261

1.0 INTRODUCTION

By letter dated May 10, 2002, as supplemented by letters dated March 12, 2003, April 10, 2003, March 5, 2004, and July 22, 2004, the Carolina Power & Light Company (CP&L, the licensee) submitted a request for changes to the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2), Technical Specifications (TS). The requested changes would make the following modifications to the HBRSEP2 TS:

> The definition of Dose Equivalent ^{131}I in Section 1.1 would be revised to reference as the iodine dose conversion factors those listed under the "Effective" column of Table 2.1 of Federal Guidance Report 11.
>
> The reactor coolant system (RCS) operational leakage limits, stated in Limiting Condition for Operation (LCO) 3.4.13, "RCS Operational Leakage," for total primary-to-secondary leakage through the steam generators would be reduced from 1 gpm to 0.3 gpm. In addition, the allowable primary-to-secondary leakage for any one steam generator would be reduced from 500 gpd to 150 gpd.
>
> The maximum allowable activity levels of Dose Equivalent ^{131}I in TS 3.4.16, "RCS Specific Activity," would be reduced from 1µCi/g in Condition A and in Surveillance Requirement 3.4.16.2 to 0.25 µCi/g. In addition, Figure 3.4.16-1, "Reactor Coolant Dose Equivalent ^{131}I Specific Activity Level Versus Percent of Rated Thermal Power," would be deleted. Required Action A.1 would be revised to replace the reference to the acceptable region of Figure 3.4.16-1 with a limit of $\leq$ 60 µCi/g Dose Equivalent ^{131}I. The second entry condition of Condition C would be revised to replace the reference to the unacceptable region of Figure 3.4.16-1 with a reference to >60 µCi/g Dose Equivalent ^{131}I.

The licensee withdrew a proposal to revise the description of the Explosive Gas and Storage Tank Radioactivity Monitoring Program in TS 5.5.12. The licensee had also proposed a change to Appendix B, "Additional Conditions," related to cycle length restriction. This aspect is no longer a part of this amendment request as this restriction was removed by Amendment No. 200 issued by the NRC on March 10, 2004.

The licensee proposed these revisions to the TS as a result of its application for a full-scale implementation of the alternative source term (AST) under 10 CFR 50.67.

The April 10, 2003, March 5, 2004, and July 22, 2004, letters provided clarifying information that did not expand the scope of the proposed amendment as described in the original notice of proposed action published in the *Federal Register* and did not change the initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

A licensee's adoption of AST requires analyses of those accidents appropriate for the type of reactor facility. NRC guidance on the performance of such analyses is presented in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000. An acceptable demonstration involves showing that both the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) doses are less than 25 rem Total Effective Dose Equivalent (TEDE) or some fraction thereof, depending upon the accident. In addition, licensees must demonstrate that the control room operator dose meets 10 CFR 50.67 requirements. Accidents that are typically analyzed are based upon reactor type. For a pressurized water reactor, the typical accidents analyzed include the Main Steamline Break (MSLB), the locked rotor, the rod ejection, the steam generator tube rupture (SGTR), the fuel-handling accident, and the large-break Loss-of-Coolant Accident (LOCA). For the application of the AST to HBRSEP2, the licensee calculated the dose to all of these accidents except the rod ejection and the fuel-handling accident. Previously, the licensee had submitted the fuel-handling accident utilizing AST (March 13, 2002). Approval of the use of AST for the fuel-handling accident was issued for HBRSEP2 by Amendment 195, dated October 4, 2002. The licensee did not calculate doses for a rod-ejection accident for the reasons discussed in Section 3.3.5 of this Safety Evaluation. The licensee did calculate the consequences of a single rod control cluster assembly (RCCA) withdrawal.

The licensee calculated doses for individuals located at the EAB, the LPZ, and for individuals located in the control room and the Technical Support Center (TSC)/Emergency Operating Facility (EOF). As part of the licensee's implementation of the AST, the licensee also calculated new onsite and offsite atmospheric dispersion values.

A licensee's implementation of AST may necessitate changes to a facility's TS. For HBRSEP2, those TS that were proposed for change included the definition for Dose Equivalent ^{131}I, the operational leakage limits for the RCS primary-to-secondary, and the maximum allowable activity levels of Dose Equivalent ^{131}I in the RCS.

3.0 TECHNICAL EVALUATION

3.1 Definition of Dose Equivalent ^{131}I

The licensee proposed a change in the definition of dose equivalent ^{131}I. CP&L proposed to define dose equivalent ^{131}I using the "Effective" Column from Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion."

The NRC staff has reviewed the proposed revision to the definition and finds the licensee's proposal acceptable.

3.2 Explosive Gas and Storage Tank Radioactivity Monitoring Program

The licensee originally proposed to change the criterion from 500 mrem whole body to 500 mrem TEDE. The NRC staff indicated that the appropriate TEDE criterion for the release of the contents of a waste gas decay tank is 100 mrem TEDE. The 100 mrem value is consistent with limits of 10 CFR 20.1301 and the guidance in Branch Technical Position (BTP) ETSB 11-5. Licensees are not required to change the criterion to a TEDE as a part of the full implementation of AST. They may maintain their existing criterion at 500 mrem whole body. But if they wish to change to a TEDE criterion, the criterion must be 100 mrem TEDE. In its March 5, 2004, letter, the licensee chose to withdraw its request rather than change the dose criterion to 100 mrem TEDE. Consequently, the acceptable dose criterion will remain at 500 mrem whole body for the release of the contents of the waste gas decay tank.

3.3 Assessment of Radiological Consequences

3.3.1 Large-Break LOCA

The NRC staff performed confirmatory calculations of the potential consequences of a LOCA based upon information provided in the licensee's May 10, 2002, and March 12, 2003, submittals. The NRC staff calculated EAB, LPZ and control room operator doses. The NRC staff's calculations could not confirm that HBRSEP2's implementation of AST for the LOCA resulted in doses that met the 10 CFR 50.67 dose acceptance criteria for the EAB. This occurred because only a fraction of the total forced flow of 130,000 cfm could be justified as a mixing rate between the sprayed and unsprayed regions. The licensee confirmed the accuracy of the NRC staff's conclusion. In addition, the licensee also identified a nonconservative error in the modeling of containment leak rate.

In order to compensate for the impact of these two items on the calculated dose consequences, the licensee reevaluated and revised certain LOCA dose inputs. The changes were presented in the licensee's March 5, 2004, submittal, and related to the modeling of the containment and to the removal mechanisms in containment. There were no changes to the modeling of the dose contribution from Engineered Safety Features (ESF) leakage outside containment. There were no changes to the control room and TSC design inputs, and there were no changes to the assumptions involving atmospheric dispersion factors or core inventory source terms. Because of the change in containment modeling, the amount of activity in containment was altered, which affected the quantity of activity released from containment and the direct dose in the control room and TSC as a result of direct shine and plume shine. The licensee reevaluated these direct doses. The specific changes to containment modeling were the following:

a) the sprayed and unsprayed volumes in containment;
b) the origin of the source term associated with containment leakage;
c) the aerosol spray removal model;
d) the natural deposition removal cut-off time;
e) the mixing rate between the sprayed and unsprayed containment regions; and
f) spray removal cut-off times.

The volume of the sprayed area of the containment was originally 52 percent. It was originally postulated that only gravity affected the spray coverage patterns. In the May 10, 2002, submittal, the licensee had neglected any spray coverage occurring due to air movement patterns. In its March 5, 2004, letter, the licensee revised its spray volume calculation based upon air movement patterns described in NUREG/CR-4102, "Air Currents Driven by Sprays in Reactor Containment Buildings." As a result of this revised calculation, the spray volume of the containment was determined to be 82.9 percent with spray train A operating and 81.5 percent with spray train B operating.

In the licensee's original calculation of the LOCA consequences, it had assumed that all of the containment leakage originated from the unsprayed region of the containment. During its reassessment of the consequences of a LOCA, the licensee determined that this assumption had been implemented incorrectly into the calculation model. The licensee's revised analysis in its March 5, 2004, submittal assumed that containment leakage occurred from both the sprayed and unsprayed regions in proportion to the volumes of each region.

The licensee's original analysis in its May 10, 2002, submittal included a particulate spray removal coefficient based upon a model specified in Standard Review Plan Section 6.5.2. The licensee's March 5, 2004, letter incorporated an analysis that utilized the 50-percentile Powers spray removal model as incorporated into the RADTRAD code. This model represented a best estimate or mean removal rate values expected from mechanistic models for any given set of containment spray parameter inputs.

In the licensee's original analysis, it had assumed that natural deposition removal would cease when a decontamination factor (DF) of 1000 was achieved. The revised calculation assumed that natural deposition occurred throughout the duration of the accident.

The assumption of mechanical mixing in the containment was modified in the March 5, 2004, analysis. Originally, it was assumed that 130,000 cfm was removed from the sprayed region and transferred to the unsprayed region and 130,000 cfm was removed from the unsprayed region and transferred to the sprayed region. The revised calculation assumes that 65,000 cfm is taken from the sprayed region and transferred to the unsprayed region and visa versa.

The original calculation assumed that elemental spray removal was cut off at t = 2.46 hours when a DF of 200 was achieved. With the changes in the sprayed and unsprayed volumes and mixing rates, the time at which an elemental iodine DF of 200 was achieved became 2.08 hours for spray train A and 2.11 hours for spray train B.

The May 10, 2002, analysis established a time (20.8 hours) for particulate iodine at which a DF of 50 would be achieved. At that time, the spray removal coefficient for iodine was to be reduced by a factor of 10. With the use of the time-dependent Powers spray removal mode in the March 5, 2004, analysis, it was not necessary to apply a factor of 10 reduction when the DF equaled 50. In both the previous and the revised licensee's calculations, the sprays were secured before a DF of 50 was ever achieved.

The licensee calculated the potential consequences of a postulated large-break LOCA to the control room operators and to individuals located offsite at the EAB and at the LPZ. It was postulated that the occurrence of a LOCA would result in releases to the environment from containment leakage and emergency core cooling system (ECCS) recirculation loop leakage. The release of alkali metals and the elemental and particulate forms of iodine to containment would be reduced by containment sprays and by natural deposition when the sprays were not in operation. Containment fans would mix the radioactivity between the sprayed and unsprayed regions.

The licensee's calculations assumed containment leakage occurred at the maximum allowable leakage value in the TS containment leakage program. The containment leakage value was reduced to 50 percent of the TS value at 24 hours following the accident.

The licensee assumed ECCS leakage was twice the limit (2 gph) in the HBRSEP2 Technical Requirements Manual Specification 3.23, "Post Accident Recirculation Heat Removal System Leakage." This leakage was presumed to begin at the earliest time that recirculation flow starts and end at the time recirculation flow ceases. The licensee stated that the results of its analyses demonstrated that, in all cases, 10 CFR Part 50.67 acceptance criteria for doses were met.

Appendix A of Regulatory Guide 1.183 contains a footnote that indicates the manner in which the spray removal rate constants developed by the use of the Powers' model [i.e., the model described in NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays"] are to be selected. For design-basis calculations, these constants are to be selected in a manner that maximized the dose consequences. The licensee's selection of the 50th percentile value of the Powers model did not maximize the dose consequences. In addition, the licensee did not provide sufficient information for the NRC staff to conclude that the use of the 50th percentile value was an acceptable alternative. Therefore, the NRC staff could not approve the licensee's proposed use of the AST for the LOCA.

3.3.2 Main Steamline Break

The licensee evaluated the consequences of an MSLB. Three cases were assessed. The first case assumed the accident occurred following an iodine spike, referred to as the pre-existing spike case. The second case assumed the MSLB initiated an iodine spike, referred to as the accident-initiated spike. The third case assumed that the MSLB induced fuel failures. In all cases, a 150 gpd primary-to-secondary leak rate was assumed to the steam generator with the steamline break (referred to as the faulted steam generator). It was also assumed that the primary-to-secondary leak to the unaffected steam generators (referred to as the intact steam generators) was the remaining primary-to-secondary leakage allowed by TS, i.e., 0.19 gpm (0.3 gpm - 150 gpd).

The preexisting iodine spike case assumed the reactor coolant activity level was at the TS limit of 60 µCi/gm of dose equivalent ^{131}I. The accident-initiated spike case assumed that the reactor coolant activity level was at twice the proposed TS 3.4.16 limit of 0.25 µCi/gm dose equivalent ^{131}I. For this RCS activity level, it was assumed that, concurrent with the MSLB, an iodine spike occurs that results in the release of iodine from the fuel to the reactor coolant. Iodine was assumed to be released at a rate that is 500 times the normal iodine release rate associated with a reactor coolant activity level of 0.5 µCi/gm dose equivalent ^{131}I. The spike was assumed to occur for 8 hours. For the fuel failure case, it was assumed that two fuel assemblies were breeched and that these assemblies had a maximum radial peaking factor of 1.8. For all cases, the secondary system activity was assumed to be at the TS limit of 0.1 µCi/gm dose equivalent ^{131}I.

In all cases, steam generator dryout was assumed for the faulted steam generator. Primary-to-secondary activity released to the faulted steam generator was presumed to be released directly to the environment. Primary-to-secondary leakage to the intact steam generators was presumed to be mixed with the bulk liquid in the steam generators. The activity in the intact steam generators was assumed to be released to the environment as vapor based upon the steaming rate and the partition coefficient of the particular nuclide group. Primary-to-secondary leakage was assumed to continue until the primary-side pressure was reduced below that of the secondary side or until the temperature of the leakage was reduced below 212^{o}F. For HBRSEP2, the analysis incorporated 53.2 hours to initiate residual heat removal operation and 98.8 hours to terminate primary-to-secondary leakage.

The NRC staff performed confirmatory calculations of the potential consequences of an MSLB accident based upon information provided by the licensee. The assumptions that form the basis for the NRC staff calculations are presented in Table 3.3-1. The NRC staff calculated EAB, LPZ, and control room operator doses. The results of these calculations are presented in Table 3.3-5. The NRC staff's calculations confirmed that the implementation of AST for the MSLB at HBRSEP2 does not result in doses that exceed the acceptance criteria in Regulatory Guide 1.183.

In reviewing the licensee's analysis, the NRC staff determined that the licensee had utilized ICRP 30 dose conversion factors for determining the dose equivalent ^{131}I in primary coolant. With the proposed change in the definition of dose equivalent ^{131}I, utilization of ICRP 30 dose conversion factors is no longer appropriate. Subsequent analyses of the MSLB should be performed by calculating curie content in primary and secondary coolant based upon the revised definition of the dose equivalent ^{131}I. The effective dose conversion factors in Table 2.1 of Federal Guidance Report 11 should be used.

3.3.3 Steam Generator Tube Rupture

The licensee evaluated the consequences of an SGTR. Two cases were assessed. The first case assumed the accident occurred following an iodine spike, referred to as the preexisting spike case. The second case assumed the SGTR initiated an iodine spike, referred to as the accident-initiated spike. For the steam generators without the tube rupture, referred to as the intact steam generators, primary-to-secondary leakage was assumed to be 150 gpd per steam generator for both the preexisting and the accident-initiated spike cases. The remaining primary-to-secondary leakage allowed by TS 0.08 gpm (0.3 gpm - [2 steam generators x 150 gpd/sg]) was to the steam generator with the tube rupture (referred to as the faulted steam generator).

For the preexisting iodine spike case, it was assumed that the reactor coolant activity level was at the TS value of 60 µCi/gm dose equivalent ^{131}I. For the accident-initiated spike case, it was assumed that the reactor coolant activity level was at the proposed TS 3.4.16 limit of 0.25 µCi/gm dose equivalent ^{131}I. In the licensee's May 10, 2002, letter CP&L had indicated, "For the accident-induced iodine spike case, a similar assumption is made with one exception. The primary coolant iodine activity increases during the first eight hours of the transient as a result of the release from the defective fuel at rate 335 times the iodine equilibrium appearance rates consistent with an initial dose equivalent (DE) I-131 concentration twice the value of the proposed TS 3.4.16 limits." Based upon this statement, it would be expected that the licensee incorporated into HBRSEP2's SGTR analysis a release rate from the fuel that equated to an initial primary coolant activity level of 0.50 µCi/g of dose equivalent ^{131}I. This release rate is a factor of two greater than the release rate that would occur with primary coolant at 0.25 µCi/gm dose equivalent ^{131}I.

The licensee's March 5, 2004, letter indicated that the statement from the May 10, 2002, letter was incorrect and that the release rate was based upon primary coolant activity being at 0.25 µCi/g of dose equivalent ^{131}I. The licensee indicated that the SGTR would not result in any fuel failures.

For both the preexisting and accident-initiated spike cases, secondary system activity was assumed to be at the TS limit of 0.1 µCi/gm dose equivalent ^{131}I. In addition, for both cases, the licensee assumed the faulted steam generator would be isolated within 30 minutes and that offsite power was lost. Thus, the main condenser was unavailable for steam dump. The licensee's justification for the 30-minute assumption for isolation was that assumption was part of their current licensing basis. The licensee indicated that the original plant licensing basis established the 30-minute steam generator isolation time for the SGTR event and that this basis was reaffirmed through the NRC's review of the steam generator replacement and power uprate licensing changes in the early 1980's. Neither break flow nor primary-to-secondary leakage is assumed to be terminated within those 30 minutes.

The licensee indicated in its March 5, 2004, submittal that simulator experience has shown that the continuing break flow will not result in the opening of any main steam safety valves nor does it result in a steam generator overfill condition. The March 5, 2004, submittal also indicated that isolation of the affected steam generator was consistent with current operating procedures at HBRSEP2. In a July 22, 2004, letter that supplemented the March 5, 2004, response, the licensee indicated that it had run an SGTR scenario in June 2004 on the HBRSEP2 simulator. This scenario included a 695-gpm primary-to-secondary leak rate due to a tube rupture with an open power-operated relief valve (PORV) on the steam generator with the tube rupture and a loss of offsite power. When the open PORV was manually closed, the steam generator with the tube rupture was isolated from the environment in 20 minutes compared to the 30 minutes assumed in the AST scenario. During this scenario, the appropriate response and mitigation procedures were followed during the simulator exercise.

For both the preexisting and the accident-initiated spike cases, it was assumed that a portion of the ruptured tube flow would flash to steam. The portion that flashes to steam was assumed to rise through the bulk water in the steam generator and to enter the steam space where it would be immediately released to the environment without mitigation, i.e., no credit for scrubbing within the bulk water. Primary-to-secondary leakage to the faulted steam generator and that portion of the tube rupture flow that does not flash was assumed to mix with the bulk water and to be released to the environment based upon the steaming rate and the partition coefficient for the particular nuclide group. For intact steam generators, the activity released to the environment was also based upon the steaming rate and the partition coefficient for the particular nuclide group. Primary-to-secondary leakage would continue until either the primary-side pressure was reduced below that of the secondary side or until the temperature of the leakage was reduced below 212°F. The release of radioactivity was assumed to continue until the residual heat removal system was placed in operation. The licensee assumed that it took 53.2 hours to cool the reactor down to a point where no further release of steam and radioactivity would occur to the environment. At that time, releases from the steam generators would terminate.

Item 14 for SGTR analysis in Attachment II of the May 10, 2002, submittal stated, "The ratio of radioiodines to other radionuclides provide in the UFSAR [Updated Final Safety Analysis Report], Table 11.1.1-2, is assumed to be a constant." Based upon this statement, it could be assumed that for the preaccident spike case, where the dose equivalent ^{131}I activity level is assumed to have spiked to 60 µCi/g, a comparable spike has occurred for the other radionuclides such as cesium. This would also apply to the accident-initiated spike case. The licensee's consequences analysis for the SGTR did not appear to maintain these ratios when considering spikes. The licensee was requested to clarify their assumptions for the SGTR. The licensee provided such clarification in the March 5, 2004, letter. CP&L indicated that the statement on ratios only applies with regard to the reactor coolant activity level at the TS values (60 µCi/g or 0.25 µCi/g). In regard to spiking, only the iodine isotopes were considered to spike consistent with Appendix F of Regulatory Guide 1.183. The licensee assessed the potential for the spiking of other isotopes in addition to iodine. It concluded that some cesium spiking would occur and it would be in the form of CsI and that it would be in the particulate form. It concluded that very little of this activity would be released from the secondary side of the steam generators, especially if the steam generator tubes remain covered with water. The SGTR analysis for HBRSEP2 indicates that the tubes will remain covered in the event of an SGTR.

The NRC staff has performed their assessment of the potential consequences of an SGTR event. The NRC staff's assessment assumed that the reactor coolant activity level for dose equivalent ^{131}I was at 60 µCi/g for the preexisting spike case and at 0.25 µCi/g for the accident-initiated spike case. Table 3.3-2 presents the assumptions utilized by the NRC staff in their assessment. The potential consequences of an SGTR accident are presented in Table 3.3-5. The NRC staff's calculations confirm that the consequences of an SGTR accident met the dose criteria (25 rem TEDE preexisting spike case and 2.5 rem TEDE accident-initiated spike case) of Regulatory Guide 1.183 at the EAB.

In reviewing the licensee's analysis, the NRC staff determined that the licensee's calculations had utilized ICRP 30 dose conversion factors for determining the dose equivalent ^{131}I in primary coolant. With the proposed change in the definition of dose equivalent ^{131}I, utilization of ICRP 30 dose conversion factors is no longer appropriate. Subsequent analyses of the SGTR should be performed by calculating curie content in primary and secondary coolant based upon the revised definition of the dose equivalent ^{131}I. The effective dose conversion factors in Table 2.1 of Federal Guidance Report 11 should be used.

3.3.4 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

The licensee assessed the consequences of a postulated reactor coolant pump shaft seizure (locked rotor) event. The occurrence of such an event could result in fuel failures. In the event of fuel failures, the radioactivity from the fuel would be dispersed to reactor coolant. As a result of primary-to-secondary leakage, radioactivity would be transferred to the secondary side of the steam generator. Since it is presumed that the event will occur with a subsequent loss of offsite power, activity from the secondary side of the steam generators will be released to the environment via the process of removing the reactor's decay heat using the steam generator PORVs.

The licensee assumed that the sources of radioactivity in reactor coolant would be the operating reactor coolant activity levels assumed to be at the TS values for dose equivalent ^{131}I, the iodine spike contribution, and gap activity due to fuel failures. The primary-to-secondary leakage was assumed to mix instantaneously and homogeneously within the secondary side without flashing. The licensee's analysis consisted of a determination of (1) that fraction of fuel that reaches or exceeds the initiation temperature of fuel melt, and (2) that fraction of fuel elements for which the fuel clad is breeched. The licensee relied upon departure from nucleate boiling ratio (DNBR) as the fuel damage criterion for estimating fuel damage for the purpose of establishing releases. The licensee's analysis assumed 17 breeched assemblies. The licensee's submittal indicated that fuel melting would not occur during a locked rotor event.

Releases to the environment were assumed to occur until shutdown cooling was initiated, thereby terminating releases from the steam generators. Activity released from the steam generators would be a function of the steaming rate and the partition coefficient for the particular nuclide group.

The NRC staff performed confirmatory calculations of the potential consequences of a locked-rotor accident based upon information provided in the licensee's submittals and in response to NRC staff questions. The assumptions that form the basis for the NRC staff calculations are presented in Table 3.3-3. The NRC staff calculated EAB, LPZ, and control room operator doses. The results of these calculations are presented in Table 3.3-5. The NRC staff's calculations confirmed that the implementation of AST for the locked rotor accident at HBRSEP2 does not result in doses that exceed the acceptance criteria in Regulatory Guide 1.183.

3.3.5 Single RCCA Withdrawal

Appendix H of Regulatory Guide 1.183 provides guidance on the assessment of rod ejection type of accidents. The licensee did not perform an assessment of the consequences of a rod ejection accident. CP&L indicated that such an accident at HBRSEP2 does not result in fuel damage and the consequences are bounded by the consequences of other accidents.

The licensee did perform an assessment of the consequences of a postulated single RCCA withdrawal accident. The amount of radioactivity released as a result of this accident was based upon the number of fuel rods breeched, the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting, and the radial peaking factor.

The licensee's analysis assumed one fuel assembly had its rods breeched. Three other assemblies were assumed to reach or exceed the initiation temperature for fuel melt. The degraded fuel assemblies were assumed to release various quantities of isotopes to reactor coolant based upon the extent of the fuel damage. Radioactivity from reactor coolant would enter secondary coolant as a result of primary-to-secondary leakage. Radioactivity in the secondary-side coolant would be released to the environment based upon the steaming rate and the partition coefficient for the particular nuclide group. Primary-to-secondary leakage was assumed to continue until the primary-side pressure was reduced below that of the secondary side. The release of radioactivity from the steam generators was assumed to be terminated when the residual heat removal system was placed in operation.

In Appendix H of Regulatory Guide 1.183, two release paths to the environment were considered. One pathway is via the secondary side through the steam generator PORVs. This pathway is analyzed because all activity is assumed to be released and remain in the RCS. Therefore, the only release path is via the secondary system. For this case, no releases from the containment are assumed.

The second pathway is via containment leakage. This pathway occurs because the rod ejection is postulated to result in a loss of the RCS barrier. For this situation, 100 percent of the activity released from the fuel is assumed to be released into the containment. This would result in the pressurization of the containment to well above normal operating conditions due to the small- to medium-size LOCA. Therefore, Appendix H of Regulatory Guide 1.183 specifies that releases due to containment leakage are also to be modeled for a rod ejection accident. These two pathways are to be analyzed separately, and there is no summation of the dose results.

The licensee’s analysis did not incorporate a containment release pathway nor did it use the guidance of Appendix H. The licensee’s analysis utilized the model in Appendix G (Locked Rotor Accident) of Regulatory Guide 1.183.

The licensee concluded that analysis of the containment leak path for the RCCA withdrawal was unnecessary because primary coolant boundary is expected to remain intact and the only leakage from the RCS to containment would be minor and would not result in a pressurization of the containment. Therefore, any release from containment would be insignificant. Consequently, the licensee assumed that the only pathway necessary for consideration for release to the environment would be the primary-to-secondary leak path and the resultant release via steaming through the steam generator PORVs.

The NRC staff performed confirmatory calculations of the potential consequences of an RCCA withdrawal accident based upon information provided in the licensee’s submittals and in response to NRC staff questions. The NRC staff also assessed the licensee’s position that the containment pathway did not need to be evaluated for the RCCA withdrawal accident. The NRC staff’s assessment concluded that the licensee did not need to analyze an RCCA withdrawal accident assuming a containment leak pathway. The assumptions that form the basis for the NRC staff calculations are presented in Table 3.3-4. The NRC staff calculated EAB, LPZ, and control room operator doses. The results of these calculations are presented in Table 3.3-5. The NRC staff’s calculations confirmed that the implementation of AST for the locked rotor accident at HBRSEP2 does not result in doses that exceed the acceptance criteria in Appendix H of Regulatory Guide 1.183.

3.4 Assessment of Control Room Habitability

The licensee calculated the control room operator dose for the accidents evaluated in Section 3.3 of this Safety Evaluation. When operated to mitigate the consequences of an accident, the control room emergency ventilation system brings 400 cfm of outside air, passes it through a filter and a charcoal adsorber, and distributes the air to the control room envelope. Air from the control room envelope is recirculated back through the filter and the charcoal adsorber at a rate of 2600 cfm.

The control room emergency ventilation system is actuated either by a safety injection signal or a signal from a radiation monitor. The licensee's analyses for a locked rotor event and for a single RCCA withdrawal assumed that the control room normal ventilation system operated for 1 hour prior to switching to the control room's emergency filtration system. For the SGTR accident, it was assumed that the control room emergency ventilation system was initiated 310 seconds following the start of the accident. During the period when the normal control room ventilation system is operating, it is assumed that the normal makeup air to the control room envelope is 400 cfm, which is unfiltered, and that the unfiltered inleakage into the envelope is 300 cfm.

During the periods when the control room's emergency filtration system is operating, the licensee's analyses assumed that unfiltered inleakage into the control room envelope is initially 300 cfm for all accidents except the LOCA. For the LOCA, it is assumed that the unfiltered inleakage is initially at 170 cfm. After 1 hour, the unfiltered inleakage is assumed to be reduced by 70 cfm when the inleakage from the Hagan Room is reduced. The Hagan Room is a source of unfiltered inleakage into the control room envelope when there is a loss of the Auxiliary Building exhaust fan HVE-7. Loss of this fan results in the Hagan room, which is adjacent to the control room envelope, being at a higher pressure than the control room envelope. The Hagan Room will remain in this condition until operators can take certain actions which will result in the reduction of the pressure in the Hagan Room to below that of the control room envelope. The licensee's analysis assumed that it would take approximately one hour to implement the actions and reduce the pressure in the Hagan Room.

The licensee has performed testing of its control room envelope to establish its inleakage characteristics. A summary of these testing results were presented in an April 10, 2003, letter to the NRC. In this letter, the licensee indicated that the inleakage test acceptance criteria are less than 560 cfm (400 cfm unfiltered makeup plus 160 cfm of unfiltered inleakage) when in the normal operating mode; 160 cfm when the control room emergency filtration system is operating in the emergency pressurization mode and the Hagan Room is at a greater pressure than the control room envelope; and less than 90 cfm when the Hagan Room is at a lesser pressure than the control room envelope. The licensee did test the control room envelope to determine its inleakage characteristics when the normal control room ventilation system was operating. The NRC staff will review the licensee's response to Generic Letter 2003-01, "Control Room Habitability," to determine the manner in which the licensee confirmed the inleakage.

The NRC staff performed calculations to determine whether the licensee's implementation of the AST would result in postulated doses that would meet 10 CFR 50.67. The NRC staff's calculations confirmed that control room operators' doses do meet 10 CFR 50.67 for the accidents discussed in Sections 3.3.2 through 3.3.5.

3.5 Atmospheric Relative Concentration Estimates

3.5.1 Meteorological Data

CP&L calculated new relative concentration (X/Q) values for the design-basis accident (DBA) dose assessments described above using onsite meteorological data collected during calendar years 1988 through 1996. These data were previously evaluated and are discussed in the Safety Evaluation associated with Amendment No. 195 dated October 4, 2002.

3.5.2 EAB and LPZ Relative Concentration Estimates

The licensee calculated X/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 Regulatory Guide 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.” The licensee made calculations for an EAB distance of 425 meters and LPZ distance of 7242 meters. Releases were assumed to be ground level. The licensee provided EAB X/Q estimates for time periods of longer than a 2-hour duration, but such estimates are not appropriate for use in this EAB dose assessment, and they are not approved as part of this license amendment.

3.5.3 Control Room Relative Concentration Estimates

CP&L used the ARCON96 methodology (NUREG/CR-6331, Revision 1, “Atmospheric Relative Concentrations in Building Wake”) for calculation of control room X/Q values with a modification to the surface roughness length and averaging sector width constant. These two modifications are acceptable to the NRC staff. Calculations were made for postulated DBA releases to the control room from the plant stack, closest main steam safety valve/relief valve, closest main steamline, nearest point of the containment building, and residual heat removal heat exchanger room, and to the TSC/EOF from the nearest point of the containment building and residual heat remover heat exchanger room. All releases were assumed to be ground-level point releases.

3.5.4 Summary - Atmospheric Relative Concentration Estimate Analysis

The NRC staff has reviewed the inputs to the PAVAN and ARCON96 codes and found them to be generally consistent with NRC staff practice, site configuration drawings, and other information provided by CP&L. Although the NRC staff is of an opinion that trees may have had an influence on meteorological measurements at HBRSEP2 in the 1988 through 1996 time period, the NRC staff does not have sufficient basis for concluding that the impact is significant enough to reject the dose assessment for this amendment given the assumptions used in the calculations. Based on this review, the NRC staff finds the X/Q values listed in Table 3.5-1 acceptable for use in this dose assessment.

4.0 SUMMARY

The Commission has concluded, based on the considerations discussed above, that the AST amendment request can be approved for all of the above-noted accidents except the LOCA, and that the results of the above-noted accident analyses confirm that the proposed TS changes to re-define Dose Equivalent ^{131}I, to change the allowable levels of Dose Equivalent ^{131}I in primary coolant, and to change the allowable steam generator primary-to-secondary leak rates are acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of South Carolina 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 the 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 (68 FR 15758). 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: L. Brown
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Dated: September 24, 2004

Attachments: Tables

Table 3.3-1 Assumptions for MSLB Accident

Parameter	Value
Iodine & Alkali Metals Partition Factor	
Faulted Steam Generator	1
Intact Steam Generators	0.1
Steam Release from Faulted Steam Generator (lbs)	
At time of break	137,294 SG plus 23,900 Feedwater
0-2 hours	161,304 [Includes the above]
2-8 hours	330
8-24 hours	881
24-53.2 hours	1608
53.2-98.8 hours	2512
Steam Release from Intact SGs (lbs)	
0-2 hours	300,116
2-8 hours	561,235
8-24 hours	1,110326
24-53.2 hours	1,611,092
53.2-98.8 hours	0
Primary-to-Secondary Leak Rate (gpm)	
Intact SGs	0.19
Faulted SG	0.11
Primary Coolant Activity Level - Dose Equivalent ^{131}I (µCi/g)	
Pre-existing Spike	60
Accident Initiated Spike	0.50
Secondary-Side Activity (µCi/g)	
Dose Equivalent ^{131}I	0.1
Alkali Metals	0.1 of Primary Coolant Value
Number of Failed Assemblies	2
Total Number of Fuel Assemblies	157
Time before RHR Operation (hours)	53.2
Time before Primary-to-Secondary Leak Terminated (hours)	98.8

Steam Generator Mass -Minimum (lbs)	88,641
Primary Coolant System Mass - Minimum (lbs)	372,137
Maximum Nominal Letdown Flow (gpm) @130°F, 2235 psig	120
Uncertainty Applied to Letdown Flow	10%
Maximum Identified Primary Coolant Leakage (gpm)	10
Maximum Unidentified Primary Coolant Leakage (gpm)	1
Maximum Primary Coolant Mass (lbs)	433,859
Isotopic Equilibrium Appearance Rate (Ci/hr) @Spiking Factor = 500	
^{131}I	4162
^{132}I	3914
^{133}I	7766
^{134}I	4742
^{135}I	3805
^{134}Cs	440
^{137}Cs	64
^{138}Cs	2377

Atmospheric Dispersion Factors (sec/m^3)

EAB	1.77E-3
LPZ	
0-2 hours	8.92E-5
2-8 hours	3.50E-5
8-24 hours	2.19E-5
1-4 days	7.95E-6
4-30 days	1.85E-6
Control Room (Faulted SG)	
0-2 hours	2.48E-3
2-8 hours	1.57E-3
8-24 hours	7.05E-4
1-4 days	4.74E-4
4-30 days	3.93E-4

Control Room (Intact SG)	
0-2 hours	2.60E-3
2-8 hours	1.65E-3
8-24 hours	7.22E-4
1-4 days	4.97E-4
4-30 days	4.01E-4
Breathing Rates (m^3/sec)	
Offsite	
0-8 hours	3.47E-4
8-24 hours	1.75E-4
1-30 days	2.32E-4
Control Room	3.47E-4
Chemical Form of Release to Environment	
Elemental	0.97
Organic	0.03

Table 3.3-2 Assumptions for SGTR Accident

Parameter	Value
Partition Factor for Iodine & Alkali Metals	
Flashed Steam (Faulted Steam Generator)	1.0
Non-flashed (Faulted)	0.1
Intact Steam Generators	0.1
Steam Release from Faulted SG (lbs)	
0-0.5 hours	95,500
0.5-2 hours	0
Steam Release from Intact SGs (lbs)	
0-0.5 hours	104,641
0-2 hours	302,696
0-8 hours	871,641
0-24 hours	2,002,409
0-53.2 hours	3,650,872
Break Flow to Faulted SG(lbs)	
0-0.5 hours	131,000
0.5-2 hours	0
Primary-to-secondary Leak Rate (gpm)	
Intact SGs	0.22
Faulted SG	0.08
Faulted SG Isolated (min)	30
Primary Coolant Activity Level - Dose Equivalent ^{131}I (µCi/g)	
Pre-existing Spike	60
Accident Initiated Spike	0.25
Flashing Fraction	
0-30 minutes	0.3027
Duration of Plant Cooldown by Secondary System (hr)	53.2
Steam Generator Mass -Minimum (lbs)	88,641
Primary Coolant System Mass - Minimum (lbs)	372,137

Maximum Nominal Letdown Flow (gpm) @130°F, 2235 psig	120
Uncertainty Applied to Letdown Flow	10%
Maximum Identified Primary Coolant Leakage (gpm)	10
Maximum Unidentified Primary Coolant Leakage (gpm)	1
Isotopic Equilibrium Appearance Rate @Spiking Factor = 335 (Ci/hr)	
^{131}I	2081
^{132}I	1958
^{133}I	3886
^{134}I	2371
^{135}I	1907

Atmospheric Dispersion Factors (sec/m^3)

EAB	1.77E-3
LPZ	
0-2 hours	8.92E-5
2-8 hours	3.50E-5
8-24 hours	2.19E-5
1-4 days	7.95E-6
4-30 days	1.85E-6
Control Room	
0-2 hours	2.60E-3
2-8 hours	1.65E-3
8-24 hours	7.22E-4
1-4 days	4.97E-4
4-30 days	4.01E-4

Breathing Rates (m^3/sec)

Offsite	
0-8 hours	3.47E-4
8-24 hours	1.75E-4
1-30 days	2.32E-4
Control Room	3.47E-4

Chemical Form of Release to Environment

Elemental	
Organic	0.97
	0.03

Table 3.3-3 Assumptions for Locked Rotor Accident

Parameter	Value
Core Thermal Power Level (MWt)	2346
Duration of Plant Cooldown by Secondary System (hr)	53.2
Gap Fraction:	
^{131}I	0.08
^{85}Kr	0.10
Other Noble Gases & Halogens	0.05
Alkali Metals	0.12
Failed Fuel Assemblies	17
Primary-to-Secondary Leak Rate (gpm)	0.3
Iodine & Alkali Metals Partition Factor in Steam Generators	0.01
Steam Released from 3 SGs (lbs)	
0-2 hours	301,967
2-8 hours	566,768
8-24 hours	1,124,996
24-53.2 hours	1,637,910
Maximum Radial Peaking Factor	1.8
Steam Generator Mass -Minimum (lbs)	88,641
Primary Coolant System Mass - Minimum (lbs)	372,137
Atmospheric Dispersion Factors (sec/m^3)	
EAB	1.77E-3
LPZ	
0-2 hours	8.92E-5
2-8 hours	3.50E-5
8-24 hours	2.19E-5
1-4 days	7.95E-6
4-30 days	1.85E-6
Control Room	
0-2 hours	2.60E-3
2-8 hours	1.65E-3

8-24 hours	7.22E-4
1-4 days	4.97E-4
4-30 days	4.01E-4

Breathing Rates (m^3/sec)

Offsite	
0-8 hours	3.47E-4
8-24 hours	1.75E-4
1-30 days	2.32E-4
Control Room	3.47E-4

Chemical Form of Release to Environment

Elemental	
Organic	0.97
	0.03

Table 3.3-4 Assumptions for RCCA Withdrawal Accident

Parameter	Value
Core Thermal Power (MWt)	2346
Fuel Defects (No. of Assemblies)	
Clad Failure	1
Fuel Melting	3
Number of Fuel Assemblies in Core	157
Primary-to-Secondary Leak Rate (gpm)	0.3
Percent of Fuel which melts and releases activity to reactor coolant	
Noble Gases (%)	100
Iodines (%)	50
Percent of Fuel which melts and releases activity to containment	
Noble Gases (%)	100
Iodines (%)	25
Iodine & Alkali Metal Partition Factor in the SGs before and after the accident	0.01
Containment Volume (ft^3)	1,958,526
Containment Leak Rate (weight %/day)	
t = 0-1 day	
t > 1 day	0.10
	0.05
Gap Fraction:	
All Iodines	0.10
All Noble gases	0.10
Other Halogens	0.10
Alkali Metals	0.12
Time to Establish Shutdown Cooling (hours)	53.2
Steam Released from 3 SGs (lbs)	
0-2 hours	301,967
2-8 hours	566,768
8-24 hours	1,124,996
24-53.2 hours	1,637,910
Maximum Radial Peaking Factor	1.8

Steam Generator Mass -Minimum (lbs)	88,641
Primary Coolant System Mass - Minimum (lbs)	372,137
Atmospheric Dispersion Factors (sec/m^3)	
EAB	1.77E-3
LPZ	
0-2 hours	8.92E-5
2-8 hours	3.50E-5
8-24 hours	2.19E-5
1-4 days	7.95E-6
4-30 days	1.85E-6
Control Room (PORVs)	
0-2 hours	2.60E-3
2-8 hours	1.65E-3
8-24 hours	7.22E-4
1-4 days	4.97E-4
4-30 days	4.01E-4
Control Room (Containment)	
0-2 hours	4.15E-3
2-8 hours	2.74E-3
8-24 hours	1.17E-3
1-4 days	8.18E-4
4-30 days	6.74E-4
Breathing Rates (m^3/sec)	
Offsite	
0-8 hours	3.47E-4
8-24 hours	1.75E-4
1-30 days	2.32E-4
Control Room	3.47E-4
Chemical Form of Release to Environment	
Elemental	
Organic	0.97
	0.03

Table 3.3-6 Robinson Dose Consequences (TEDE)

Accident	Exclusion Area Boundary	Low-Population Zone	Control Room Operators
Main Steamline Break			
Preexisting Spike (Acceptance Criteria)	0.10 25	0.025 25	0.085 5
Accident-initiated Spike (Acceptance Criteria)	.63 2.5	0.12 2.5	.45 5
With Fuel Damage (Acceptance Criteria)	1.3 25	0.29 25	0.98 5
Steam Generator Tube Rupture			
Preexisting Spike (Acceptance Criteria)	19 25	0.97 25	2.72 5
Accident-initiated Spike (Acceptance Criteria)	2.4 2.5	0.12 2.5	.24 5
Locked Rotor (Acceptance Criteria)	0.42 2.5	0.035 2.5	1.6 5
Rod Control Cluster Assembly Withdrawal			
Secondary-Side Release Path	2.41	0.184	0.902
Containment Release Pathway			
(Acceptance Criteria)	6.3	6.3	5

Table 3.5-1 Robinson Relative Concentration (X/Q) Values

Offsite X/Q Values (s/m^3)

EAB	Limiting 2 hour interval	1.77 E-3
LPZ	0 - 2 hrs	8.92 E-5
	2 - 8 hrs	3.50 E-5
	8 - 24 hrs	2.19 E-5
	1 - 4 days	7.95 E-6
	4 - 30 days	1.85 E-6

Control Room and TSC/EOF X/Q Values (s/m^3)

Release - Receptor Pair	0-2 hrs	2-8 hrs	8-24 hrs	1-4 days	4-30 days
Plant Stack - CR	1.24E-03	8.97E-04	3.62E-04	2.58E-04	2.14E-04
Closest MSSV/RV - CR	2.60E-03	1.65E-03	7.22E-04	4.97E-04	4.01E-04
Closest Main Steam Line - CR	2.48E-03	1.57E-03	7.05E-04	4.74E-04	3.93E-04
Containment Nearest Point - CR	4.15E-03	2.74E-03	1.17E-03	8.18E-04	6.74E-04
Containment Nearest Point - TSC/EOF	1.64E-04	1.43E-04	6.49E-05	4.41E-05	3.50E-05
RHR Heat Exchanger Room - CR	7.13E-03	5.49E-03	2.29E-03	1.71E-03	1.37E-03
RHR Heat Exchanger Room - TSC/EOF	1.38E-04	1.23E-04	5.52E-05	3.78E-05	3.01E-05
FHB Wall - CR	1.34E-03	1.02E-03	4.31E-04	3.21E-04	2.56E-04

CR - Control Room
FHB - Fuel-Handling Building
MSSV/PORV - Main Steam Safety Valve/Relief Valve
RHR - Residual Heat Removal
TSC/EOF - Technical Support Center/Emergency Offsite Facility

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