May 30, 2002

Mr. J. S. Keenan
Vice President
Brunswick Steam Electric Plant
Carolina Power & Light Company
Post Office Box 10429
Southport, North Carolina 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT RE: ALTERNATIVE SOURCE TERM (TAC NOS. MB2570 AND MB2571)

Dear Mr. Keenan:

The Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 221 to Facility Operating License No. DPR-71 and Amendment No. 246 to Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant (BSEP), Units 1 and 2. The amendments are in response to your application dated August 1, 2001, as supplemented by letters dated November 28, 2001, December 17, 2001, January 24, 2002, February 4, 2002 (two letters), April 25, 2002, May 10, 2002, and May 28, 2002. The supplemental letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the initial *Federal Register* notice.

The amendments change the Technical Specifications (TS) to replace the current accident source term used in design basis radiological analyses with an alternative source term (AST) pursuant to 10 CFR 50.67, “Accident Source Term.” The amendment also adds license conditions to Appendix B of the Operating License for Unit 2.

The NRC staff previously reviewed the portion of the August 1, 2001, application that the licensee had segregated to facilitate the review related to the Fuel-Handling Accident (FHA) event. The FHA portion of the application was reviewed separately by the NRC staff to support core alterations and fuel handling activities during the recently completed refueling outage. The NRC approved TS changes in Amendment No. 218 to Facility Operating License No. DPR-71 and Amendment No. 244 to Facility Operating License No. DPR-62 for BSEP, Units 1 and 2, dated March 14, 2002. The previous amendments were associated with (1) secondary containment operability during core alterations, and (2) selective implementation of the alternative source term to the FHA. The enclosed amendments address the remainder of your August 1, 2001, application.

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

Sincerely,

/RA/

Brenda L. Mozafari, Sr. Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-325
and 50-324

Enclosures:
1. Amendment No. 221 to
License No. DPR-71
2. Amendment No. 246 to
License No. DPR-62
3. Safety Evaluation

cc w/enclosures:
See next page

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

Sincerely,

/RA/

Brenda L. Mozafari, Sr. Project Manager, Section 2
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1. Amendment No. 221 to License No. DPR-71
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cc w/enclosures:
See next page

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ACCESSION NUMBER: ML021480483 ** See previous concurrence

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CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 221
License No. DPR-71

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

   A. The application for amendment filed by Carolina Power & Light Company (the licensee), dated August 1, 2001, as supplemented by letters dated November 28, 2001, December 17, 2001, January 24, 2002, February 4, 2002 (two letters), April 25, 2002, May 10, 2002, and May 28, 2002, 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 2.C.(2) of Facility Operating License No. DPR-71 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 221, 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 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Thomas Koshy, Acting 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: May 30, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 221

FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

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
1.1-3	1.1-3

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 246
License No. DPR-62

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

   A. The application for amendment filed by Carolina Power & Light Company (the licensee), dated August 1, 2001, as supplemented by letters dated November 28, 2001, December 17, 2001, January 24, 2002, February 4, 2002 (two letters), and April 25, 2002, May 10, 2002, and May 28, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Operating License and Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-62 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 246, 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 upon completion of Unit 2 Refueling Outage 15.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Thomas Koshy, Acting Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and the Technical Specifications

Date of Issuance: May 30, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 246

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

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
1.1-3	1.1-3

Remove Appendix B of Operating License DPR-62 and replace it with the attached Appendix B. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 221 TO FACILITY OPERATING LICENSE NO. DPR-71

AND AMENDMENT NO. 246 TO FACILITY OPERATING LICENSE NO. DPR-62

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

By letter dated August 1, 2001, as supplemented by letters dated November 28, 2001, December 17, 2001, January 24, 2002, February 4, 2002 (two letters), April 25, 2002, May 10, 2002, and May 28, 2002, Carolina Power and Light Company (CP&L) requested a license amendment for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2. The amendment request proposed replacing the current accident source term used in design basis radiological analyses with an alternative source term (AST) pursuant to Title 10, *Code of Federal Regulations* (10 CFR) Section 50.67, "Accident Source Term." Several other changes to the licensing basis and changes to the technical specifications (TS) were requested. In Amendment No. 218 to Facility Operating License No. DPR-71 and Amendment No. 244 to Facility Operating License No. DPR-62 dated March 14, 2002, the U.S. Nuclear Regulatory Commission (NRC) approved the TS changes associated with secondary containment operability during refueling operations and the selective implementation of the AST to the fuel-handling accident (FHA) design-basis accident (DBA) radiological consequences analysis. The November 28, 2001, December 17, 2001, February 4, 2002 (two letters), April 25, 2002, and May 28, 2002, letters provided clarifying information only and did not change the initial proposed no significant hazards consideration determination, or expand the scope of the initial *Federal Register* notice.

This safety evaluation (SE) provides the NRC staff's review of the licensee's DBA radiological consequences analyses and proposed changes to TS that remain from the August 2001 submittal. The licensee proposes to implement an AST for the remaining DBA analyses and to use revised atmospheric dispersion factors for release points and receptors associated with the DBAs. The licensee also proposes to revise the TS to conform with the AST implementation and the assumptions used in the supporting DBA analyses. Specifically, the licensee proposes to substitute the existing reference for the definition of "Dose Equivalent Iodine-131" in Section 1.1 with Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors," 1989, and FGR-12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993. The word "thyroid" is also to be deleted. In addition, the licensee proposes to change references to 10 CFR 100 throughout the TS and Bases to instead refer to 10 CFR 50.67.

Enclosure

The licensee did not request any increase in the allowable main steam isolation valve (MSIV) leakage rate or to delete the requirements for the currently installed MSIV leakage control system. The licensee proposed to use the main steam piping, drain lines, and main condenser as an alternate means for MSIV leakage treatment. Because the original design basis of certain main steam piping and components is not Seismic Category I, the licensee has performed or, in the case of Unit 2, has committed to perform, evaluations and seismic verification walkdowns to demonstrate that the main steam system piping and components which comprise the alternate leakage treatment (ALT) system are seismically rugged and are able to perform the safety function of an MSIV leakage treatment system. By letter dated May 28, 2002, the licensee provided a license condition for its commitment to complete seismic walkdowns on Unit 2.

The licensee also performed an evaluation of the seismic adequacy of the turbine building which houses the ALT system. The structural integrity of the turbine building is an important consideration to the adequacy of the alternate MSIV leakage path because a non-seismically designed turbine building should be capable of withstanding the earthquake without degrading the capability of the ALT system.

The licensee referenced the General Electric Company (GE) Report, NEDC-31858P, Rev. 2, entitled "BWROG [Boiling Water Reactor Owners Group] Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," (BWROG Report) (Ref. 3), as a basis for the acceptability of its proposed license amendment. The BWROG report summarizes data on the seismic performance of main steam piping and condensers in past strong-motion earthquakes at various facilities, and compares design attributes of the piping and condensers with those in typical GE Mark I, II, and III nuclear plants. The NRC staff, in its SE of the BWROG report dated March 3, 1999 (Ref. 4), determined that the BWROG approach of utilizing the earthquake experience-based methodology, supplemented by plant-specific seismic adequacy evaluations is an acceptable basis to demonstrate the seismic ruggedness of non-seismically analyzed main steam system piping and condensers. However, the NRC staff identified certain limitations that required individual licensees to provide plant-specific design information and evaluation when BWROG approach was elected for resolving the MSIV leakage issue. In response to the NRC staff-identified limitations, the licensee provided a BSEP-specific design information and evaluation in Reference 1.

2.0 BACKGROUND

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

In December 1999, the NRC issued a new regulation, 10 CFR 50.67, “Accident Source Term,” which provided a mechanism for licensed power reactors to replace the traditional accident source term used in their DBA analyses with an alternative source term. Regulatory guidance for the implementation of these ASTs is provided in Regulatory Guide (RG) 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.” Section 50.67 of 10 CFR requires a licensee seeking to use an AST to apply for a license

amendment and requires that the application contain an evaluation of the consequences of affected DBAs. CP&L's application of August 2001, as supplemented, addresses these requirements in proposing to use the AST described in RG 1.183 as the DBA source term used to evaluate the radiological consequences of DBAs for BSEP Units 1 and 2. As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67 (b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, General Design Criterion-19, for the loss-of-coolant accident (LOCA), the main steam line break (MSLB) accident, and the control rod drop accident (CRDA).

3.0 EVALUATION

3.1 Seismic Ruggedness of Alternative Leakage Treatment Path

The primary components to be relied upon for the proposed ALT systems are the main steam drain lines downstream of the outboard MSIV to the isolated main condenser. Other potential drain pathways are system drains, such as high pressure coolant injection and reactor core isolation cooling system drains, which are routed to the main condenser. The condenser forms the ultimate boundary of the ALT system. Boundaries upstream of the condenser were established by utilizing existing valves, and were used to limit the extent of the seismic verification walkdown.

The licensee's evaluation included: (1) a comparison of the BSEP design-basis earthquake (DBE) ground spectrum, with the seismic ground motion spectra that were experienced at the facilities in the earthquake experience database, (2) a comparison of BSEP condenser design parameters with the condensers in the earthquake experience database, and anchorage capacity of the condenser assembly, (3) piping and pipe support data comparison of the BSEP non-seismically analyzed ALT piping with the earthquake experience database that demonstrated good seismic performance, and (4) capability of the turbine building to withstand the BSEP DBE loads.

The licensee performed, or in the case of Unit 2, will perform (pursuant to license condition), seismic verification walkdowns to demonstrate the seismic ruggedness of the ALT system and components. The ALT system and components were evaluated for the following design attributes: (1) piping, pipe support and equipment seismic vulnerabilities, such as excessive span, heavy unsupported components, non-ductile piping or support material, localized stresses, severe corrosion, and anchorage, (2) seismic anchor movement, (3) seismic interaction (i.e., II/I) and proximity, (4) valve attributes, and (5) condenser and its anchorage.

3.1.1 Reliability of Boundary Valves

The licensee stated that under normal operating conditions, a flow path is in service to the main condenser through flow orifices. The orifice bypass valves (i.e. MS-038A through F038D and MVD-F021) are normally closed. When required, the ALT path will be established by opening the orifice bypass valves. The orifice bypass valves are not redundant and, therefore, a backup ALT pathway has also been designed to ensure the availability of a highly reliable path. The establishment of the ALT path relies upon the opening of the five motor-operated orifice bypass valves (MS-F038A, MS-F038B, MS-F038C, MS-F038D, and MVD-F021). The licensee committed to add the five motor-operated valves to the BSEP augmented in-service testing

(IST) program, and to periodically stroke-time the valves. In addition, the licensee is required by license condition to add ALT boundary check valves (MVD-V5008 and V5009) to the BSEP plant check valve program to ensure valve reliability. Based on the above, the NRC staff believes the licensee has provided sufficient basis to ensure that the valves and the pathway will function as required to establish the primary MSIV leakage pathway to the condenser.

3.1.2 Seismic Verification Walkdown

The ALT system consists of the main steam piping (beyond the outboard MSIVs), the steam drain lines, the condenser, and interconnected piping. The ALT system, in general, is not seismically analyzed since such analysis was not required in the original licensing basis of either unit at Brunswick.

In order to confirm the functional capability of the ALT system, the licensee performed seismic verification walkdowns for Brunswick Unit 1. The purpose of the walkdowns was to ensure that the ALT system falls within the bounds of the design characteristics of the seismic experience database as discussed in Section 6.7 of the BWROG Report. Specifically, the walkdowns were performed to (1) verify that BSEP features have attributes similar to those in the earthquake experience database that have demonstrated good seismic performance, (2) verify general conformance of BSEP pipe support spans to the requirements of USA Standards (USAS) B31.1.0-1967, and (3) examine the BSEP ALT system from the outboard MSIVs to the condenser to identify potential seismic vulnerabilities considering those structural details and causal factors that resulted in component damage at database plants.

The walkdowns focused on piping systems that were not seismically analyzed in the original design. The potential vulnerabilities which were identified as "outliers" include categories such as support failure, failure of non-seismically designed plant features (II/I), proximity and impact, and differential seismic anchor motion on piping systems. The licensee provided a list of outliers identified during the walkdowns. The licensee evaluated and analyzed the outliers to demonstrate acceptability as-is, or to implement plant modifications to resolve the walkdown findings. As a result of the walkdowns and the subsequent evaluations, the licensee identified the need for the following modifications:

Unit 1

(1) Main Steam Drains to Condenser: add a new dead load support,

(2) Main Steam Drip Leg Drains: add a new support, and

(3) MSR/RFP/SJAE Drains: add spring check valves to isolate RFP, MSR, and SJAE drains.

Unit 2

The licensee indicated that BSEP, Unit 2, seismic verification walkdown will be performed during the Unit 2 Refueling Outage 15 and committed to resolve identified outliers by analysis and/or modifications during the outage scheduled to begin in March 2003. This commitment is addressed by adding a license condition as requested by letter dated May 28, 2002.

The licensee has committed to complete all modifications prior to the restart of each unit of the plant from its respective upcoming refueling outages.

3.1.3 Comparison of BSEP and Experience Data

In order to demonstrate the applicability of the BWROG's earthquake experience-based methodology to equipment at BSEP, the licensee provided a comparison between earthquake experience database sites spectra and BSEP, Units 1 and 2, DBE ground spectrum. The comparison of the spectra shows that the database sites ground motions envelop the BSEP, Units 1 and 2, DBE ground spectrum within the frequency range of 1 Hz to 33 Hz. On the basis of this comparison, the NRC staff concludes that the BSEP, Units 1 and 2, DBE ground spectrum is bounded by the earthquake experience database sites spectra in the frequency range of design interest. The licensee's use of the BWROG's earthquake experience-based methodology was reviewed by the NRC staff and found acceptable to verify the seismic adequacy of equipment in the alternative MSIV leakage pathway.

Based on the NRC staff's review of the information provided in the licensee's submittal of September 27, 2001, the NRC staff requested additional information related to piping and pipe supports. The licensee provided its response to the NRC staff's request for additional information in its submittal of January 24, 2002 (Ref. 2). In Ref. 1, the licensee provided a database for main steam and process piping at: Valley Steam Plant Units 1 and 2; El Centro Steam Plant; Humboldt Bay Unit 3; and Moss Landing Power Plant Units 1, 2, 3, 4, 5, 6, and 7. The piping attributes comparison included pipe diameter, pipe schedule, pipe wall thickness, and pipe diameter-to-thickness (D/t) ratio. The licensee also provided a data comparison between the BSEP ALT system piping and the piping of the above selected database facilities. The NRC staff reviewed the analyses and concurs with the licensee's judgment that the BSEP ALT piping data are enveloped by the experience database.

In response to the NRC staff's request for additional information, the licensee stated that the BSEP ALT system piping conservatively used the U-pipe support spacing suggested in USAS B31.1.0-1967 as screening guidelines for support spacing to ensure that the BSEP ALT system piping spans are bounded by those typically found in the earthquake experience database sites. The licensee also stated that the typical BSEP pipe support configurations are the same as those typically found in the earthquake experience database sites. The NRC staff determined that the comparison of the earthquake experience database with the corresponding BSEP ALT piping is adequate and is, therefore, acceptable.

In Reference 1, the licensee also compared the structural characteristics of the BSEP main condenser to those of similar database condensers that have experienced significant earthquakes as addressed in the BWROG Report. The BSEP condenser is a single-pass, single-pressure surface condenser. The licensee compared design attributes of the BSEP condenser with the database condensers from Moss Landing, Units 6 and 7. For the BSEP condenser, the operating weight is smaller and the height is lower than the database condensers. The licensee stated that the database condensers have physical arrangements and construction details similar to the BSEP condenser and would function similarly in their responses to seismic excitations. In response to the NRC staff's request for additional information, the licensee stated that based on its seismic DBE evaluation of the BSEP condenser, the maximum stress in the condenser shell due to combined dead load and seismic DBE load is significantly less than the Code-allowable stress limit. The licensee's calculation

also shows that the maximum shear stress across the condenser shell is very small. The licensee determined that the anchors for the BSEP condenser assembly have adequate capacity to prevent overturning due to the combined DBE seismic and operating loads.

As a result of the NRC staff's review of the licensee's description of planned actions to meet the provision of the BWROG report, the licensee provided a license condition dated May 28, 2002. The license condition commits the licensee to complete the stated actions prior to the restart of Unit 2 from the March 2003 outage. This commitment provides the necessary assurance that the ALT systems components will possess sufficient capability to withstand the combined DBE seismic and operating loads.

3.1.4 Analyses for Alternate Leakage Treatment Pathway

The NRC staff, in its SE of BWROG report (Ref. 4), required individual licensees of plants whose Final Safety Analysis Reports (FSARs) or Updated FSARs (UFSARs) reference Appendix A to 10 CFR Part 100, "Seismic and geological siting criteria for nuclear power plants," to perform a bounding seismic analysis for the ALT path piping. In its response, the licensee stated that Appendix A to 10 CFR Part 100 is not referenced in the BSEP UFSAR and therefore, a bounding seismic analysis for the ALT piping was not performed. However, the licensee has performed seismic adequacy verification of the non-seismic ALT piping and related components and supports using the earthquake experience database approach outlined in the BWROG report. The licensee's seismic verification walkdowns are intended to identify potential piping concerns and reviews of design codes and standards, piping design parameters, and support configurations to demonstrate that the piping and related supports fall within the experience database. The licensee used USAS B31.1.0-1967 Code-suggested pipe support spacing as part of its walkdown screening criteria. In response to an NRC staff's request for additional information, the licensee stated that the seismic walkdowns reviewed the piping and tubing systems and associated supports for good design and industry standard practices and to verify that they are free from known seismic vulnerabilities identified from earthquake experience data.

The licensee stated that the non-safety related piping is generally composed of welded steel spools and standard support components, and is similar to piping found in the seismic experience database. In general conformance to the recommendation of the USAS B31.1-1967, all systems are predominantly supported for dead weight utilizing single rod hangers or rod-hung trapezes, variable spring hangers, and welded steel angle trapezes or cantilever brackets with U-bolts, and the pipe supports are attached to either concrete or structural steel. In response to the NRC staff's request for additional information, the licensee performed a comparison between the BSEP pipe supports and the earthquake experience database, and concluded that the BSEP pipe supports are similar to those typically found in the earthquake experience database sites. The licensee stated that the non-seismically analyzed main steam drain and associated piping are generally bounded in diameter and D/t ratio by those installed in the earthquake experience database facilities, as evidenced in the BWROG report and the supplemental updated earthquake performance data discussed above. The licensee stated that upon completion of all necessary modifications, piping position retention and pressure boundary integrity would be maintained by the deadweight supports under normal and earthquake loading.

Based on the NRC staff's review of the licensee's descriptions stated above, the NRC staff concurs with the licensee's conclusion that the BSEP non-seismically analyzed main steam system piping and condenser, which will be used for the ALT system, are comparable to those in the earthquake experience database, and that the seismic verification walkdowns of the system and subsequent evaluations have addressed characteristics associated with the limited component damage situations observed at the database facilities as discussed in the BWROG report and accepted in the NRC staff's topical SE. Based on the review of the information provided by the licensee, the NRC staff finds that the licensee has taken reasonable measures to ensure resolution of identified outliers.

3.1.5 Anchorage for Condenser

The BSEP condenser support anchorage consists of a center-side fixed support, a center-side sliding support, and four corner supports. The fixed center-side fixed support is anchored to a sole plate at the turbine building base mat and welded to the bottom plate of the condenser. The sole plate is anchored to the base mat with six 7/8" diameter cast-in-place bolts. All the four corner supports are positively anchored by eight 1-3/4" diameter anchor bolts cast into the basemat. The center-side sliding support consists of three anchor chairs with 1-3/4" and 2" diameter bolts. The licensee compared the BSEP condenser support anchorage with the condensers in earthquake experience database and determined that the ratio of the BSEP condenser anchorage shear area to seismic demand is substantially greater than that for the selected database sites. Based on the capacities of the anchoring system, the licensee determined that the overall anchor system of the BSEP condenser is capable of withstanding the calculated DBE loads, in combination with operating loads. Based on its review of the licensee's design information, the NRC staff finds the licensee's determination concerning the condenser anchorage seismic adequacy reasonable and acceptable.

3.1.6 Seismic Adequacy of Turbine Building

The turbine building is a seismic category II structure. The licensee performed a seismic evaluation of the BSEP turbine building to confirm that the condenser will not fail due to seismic II/I type of interaction (e.g., structural failure of the turbine building and its internals). The BSEP main condensers are located in the lower elevations of the turbine building. The turbine building is supported on spread footings and founded on structural backfill. The turbine building is of reinforced concrete construction above the main condenser and below the operating floor. Reinforced concrete shield walls are provided above the operating floor for radiation protection. The superstructure above the operating floor is a steel-framed crane bay with panel siding and roof constructed of metal deck.

The licensee stated that the turbine building is designed for both uniform building code (UBC) seismic zone 1 (0.08g) and wind loading based on fastest wind speed with a 100-year recurrence period. The licensee stated that the design of the turbine building is governed by hurricane wind load. The licensee provided a comparison of forces (lateral shear and moment) due to the design basis hurricane wind with the seismic forces based on the UBC and BSEP DBE. The comparison shows that, overall, the design basis wind lateral shear and moment forces are substantially higher than those due to seismic UBC and DBE, and therefore, the BSEP Units 1 and 2 turbine building is not expected to fail or collapse following a DBE. On the basis of the comparison of design wind load with the seismic DBE, the NRC staff concurs with the licensee's conclusion that the DBE is bounded by the design basis wind load with

substantial margin, and the turbine building is expected to remain structurally intact following a DBE to preclude condenser failure due to the seismic II/I type of interaction.

3.1.7 NRC Staff Conclusion Concerning Seismic Adequacy

Based on the above evaluation, the NRC staff concludes that upon completion of the plant modifications necessary for the identified outliers, there is reasonable assurance that the BSEP main steam lines, main steam drain lines, condenser, and associated interconnected piping and supports will be seismically adequate for the proposed MSIV ALT system. The NRC staff's conclusion is based on (1) the comparison that indicated that the BSEP DBE ground spectrum in the frequency range of design interest is well below the seismic ground motion spectra that was experienced at the facilities in the earthquake experience database, (2) the design attributes of the BSEP condenser are enveloped by the condensers in the earthquake experience database, and that the condenser assembly has sufficient anchorage capacity, (3) the majority of the main steam system piping was seismically analyzed as part of the initial design of the plant, (4) the non-seismically analyzed ALT piping is represented by the earthquake experience database that demonstrated good seismic performance, (5) the turbine building has adequate capability to withstand the DBE loads, and (6) the licensee's commitment to resolve identified outliers by analysis and/or modifications. The NRC staff, therefore, concludes with the addition of the license condition on Unit 2 that the licensee has demonstrated that the main steam system piping and components which comprise the ALT system are seismically rugged and are able to perform the safety function of an MSIV leakage treatment system.

It should be noted that the NRC staff's acceptance of the experience-based methodology as presented by the BWROG and CP&L is restricted to its application for ensuring the pressure boundary integrity and functionality of the MSIV ALT system. The NRC staff's acceptance of the methodology for this application is not an endorsement for the use of the experience-based methodology for other applications at BSEP.

As stated above, the licensee elected to utilize seismic experience-based methodology based on a GE Report, NEDC-31858P, Rev. 2, entitled "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," (BWROG Report) (Ref. 3). The NRC staff, in its SE of BWROG report (Ref. 4), determined the BWROG approach of utilizing the earthquake experience-based methodology, supplemented by plant-specific seismic adequacy evaluations, an acceptable basis to demonstrate the seismic ruggedness of non-seismically analyzed main steam system piping and condensers.

On the basis of the information provided by the licensee in References 1 and 2, the NRC staff concludes that the main steam system piping and components which comprise the ALT system are seismically rugged and are able to perform the safety function of an MSIV leakage treatment system.

3.2 Radiological Consequences of Design Basis Accidents

The NRC staff reviewed the licensee's analysis methods, assumptions, and inputs, using docketed information provided by the licensee. The licensee followed RG 1.183 guidance in performing the DBA radiological consequences analyses in support of this amendment request. The NRC staff used RG 1.183 and Standard Review Plan (SRP) Section 15.0.1 to aid in its

review of the licensee's DBA analyses. Although the NRC staff performed independent calculations as a means of confirming the licensee's results, the NRC staff's findings are based on the licensee's analyses. Table 1 provides the MSLB, CRDA, and LOCA analysis assumptions found acceptable by the NRC staff. Table 2 provides the doses projected by CP&L. Table 3 provides the RG 1.183 dose acceptance criteria for each analysis.

CP&L performed the radiological analyses that support this amendment assuming a reactor power equal to 102 percent of 2923 MWt. This power level exceeds the current licensed reactor power for BSEP, Units 1 and 2. The increased power level was used by CP&L in support of a separate extended power uprate application. This is a conservative approach for this amendment request and is acceptable to the NRC staff. Although this amendment does not revise the current licensed reactor power, this SE and that prepared for Amendment No. 218 to Facility Operating License No. DPR-71 and Amendment No. 244 to Facility Operating License No. DPR-62 will provide a basis for the NRC staff's conclusion in the separate power uprate licensing action that the radiological consequences of the proposed power uprate are acceptable.

During an October 24, 2001, meeting between the NRC staff and CP&L, the licensee was requested to provide copies of the calculations supporting the AST radiological consequence analyses. By letter dated November 28, 2001, CP&L provided these calculations, which are considered to be proprietary. These calculations included sensitivity studies to determine the bounding design basis accident scenario. The NRC staff based its determination of acceptability on those calculations that support the analyses as provided by CP&L in the submittal dated August 1, 2001, as supplemented.

3.2.1 Loss-of-Coolant Accident (LOCA)

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

3.2.1.1 Source Term

In accordance with RG 1.183 guidance, CP&L determined the inventory of fission products in the reactor core based on the uprated maximum full-power operation of the core using an appropriate isotope generation and depletion computer code. The core inventory factors (Ci/MWt) were input to the dose calculations done using the analysis code RADTRAD.[1]

Fission products from the damaged fuel are released into the reactor coolant system (RCS) and then into the primary containment (i.e., drywell and wetwell). With the LOCA, it is anticipated that the initial fission product release to the primary containment will last 30 seconds and will release all of the radioactive materials dissolved or suspended in the RCS liquid. The gap

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[1]NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Published April 1998, and Supplement 1 dated June 1999.

inventory release phase begins two minutes after the event starts and is assumed to continue for 30 minutes. As the core continues to degrade, the gap inventory release phase ends and the in-vessel release phase begins. This phase continues for 1.5 hours. Tables 1, 4, and 5 of RG 1.183 define the source term used for these two phases.

The inventory in each release phase is released at a constant rate over the duration of the phase and starting at the onset of the phase. Once dispersed in the primary containment, the release to the environment is assumed to occur through three pathways:

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

The LOCA considered in this evaluation is a complete and instantaneous severance of one of the recirculation loops. The pipe break results in a blowdown of the reactor pressure vessel (RPV) liquid and steam to the drywell via the severed recirculation pipe. The resulting pressure buildup drives the mixture of steam, water, and other gases down through vents to the downcomers and into the suppression pool water thereby condensing the steam and reducing the pressure. Due to the postulated loss of core cooling, the fuel heats up, resulting in the release of fission products. Under the traditional TID-14844 assumption of instantaneous core damage, this initial blowdown would also include fission products, a fraction of which would be retained by the suppression pool water. Under the AST, the fission product release occurs in phases over a 2-hour period. Significant quantities of fission products would not be part of the initial blowdown to the suppression pool. Subsequent recirculation of suppression pool water by the emergency core cooling system (ECCS) would cause some transport of fission products between the drywell and the wetwell, and some scrubbing effect. CP&L has conservatively assumed no credit for suppression pool scrubbing of fission products. In addition, CP&L has conservatively assumed that the fission product release from the RPV is homogeneously dispersed within the drywell free volume only, ignoring the free volume of the wetwell.

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

3.2.1.2 Suppression Pool Post-LOCA pH

The AST assumes that the iodine released to the containment includes 95 percent CsI, 4.85 percent elemental iodine, and 0.15 percent organic forms. The assumption of this iodine speciation is predicated on maintaining the suppression pool water at pH 7.0 or higher. CP&L proposes to use the standby liquid control system (SLC) to inject sodium pentaborate to the RPV, where it will mix with ECCS flow and spill over to the drywell and then to the suppression pool. Sodium pentaborate, a base, will neutralize acids generated in the post-accident primary containment environment. Credit for the SLC system in the radiological analyses is based on operation of one SLC pump, initiated 59 minutes after the event starts. The BSEP operating

procedures will be revised to direct operators, upon detection of symptoms indicating that core damage is occurring, to manually initiate the SLC system.

There is no special provision in boiling water reactors (BWRs) for controlling suppression pool pH after a LOCA. Its value will depend, therefore, on chemical species released to the suppression pool water. These species could be released from the damage core, or generated in the radiation fields existing in the containment and the drywell after a LOCA. Most of these chemical species could be 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 which is originally added to the vessel for controlling reactivity of the core.

Chemical Species Released to Suppression Pool

The licensee identified the following acidic chemicals which are introduced into the suppression pool in the post-LOCA environment: hydriodic acid, released from the damaged fuel, and nitric and hydrochloric acid formed in the containment. In addition, the suppression pool contained dissolved carbon dioxide at the concentration remaining in equilibrium with its concentration in air. The major basic chemical released from the damaged core is cesium hydroxide.

Hydriodic Acid

NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” specifies that during the gap and early in-vessel release phases, 1.5 percent of total core inventory of halogens is released as hydriodic acid. Since at the end of cycle the core contained 68.3 gm-mols of halogens, 1.025 gm-mols of hydriodic acid is released to the suppression pool water. The licensee assumed that hydriodic acid is released at a constant rate over the effective release period. Hydriodic acid is a strong acid and even at a low concentration can significantly affect suppression pool pH.

Nitric Acid

In determining the amount of nitric acid in the suppression pool, the licensee followed the methodology described in NUREG/CR-5950, “Iodine Evolution and pH Control.” This methodology assumes generation of nitric acid proportional to the integrated value of a time-dependent radiation dose rate in the suppression pool. This dose rate includes both beta and gamma radiation present in the pool. Nitric acid, in the amount of 249 gm-mols, was generated over the 30-day transient.

Hydrochloric Acid

Hydrochloric acid is generated in the post-LOCA environment by radiolytic decomposition of Hypalon cable jacketing by beta and gamma radiation. Its amount is proportional to the radiation energy absorbed by the jacketing. The methodology developed by the licensee for calculating production of hydrochloric acid was based on NUREG/CR-5950 and NUREG-1081,”Post-Accident Gas Generation from Radiolysis of Organic Materials.” The rate of production of hydrochloric acid (HCI) is given by the following equations:

$$R = R_\gamma + R_\beta$$

For γ radiation: $R_\gamma = G * \varphi_{\lambda *} S * A_\gamma$

For β radiation: $R_\beta = G * \varphi_\beta * S$

Where:
R - rate of generation of HCl
G - radiation "G" value for Hypalon (moles of HCl/unit energy absorbed)
φ - radiation flux incident to the Hypalon jacketing
S - surface of Hypalon jacketing
A_γ - absorption of γ radiation by Hypalon jacketing
(A_β is equal to 1 because it is assumed that all β radiation is absorbed by the Hypalon)

In its calculation the licensee considered the self-shielding effect of some cables and excluded some cables enclosed in conduits and totally enclosed raceways. However, it did not include reactions between HCl and metal components such as cable trays which would consume some HCl. All generated HCl was assumed to immediately dissolve in the suppression pool water. HCl generated over the 30-day transient was 3518 gm-mols.

Cesium Hydroxide

NUREG-1465 specifies that during the gap and early-in-vessel release phases, 25 percent of the cesium core inventory is released into the suppression pool. Since 28.5 percent of iodine core inventory combines with the released cesium to form cesium iodide, and the remaining cesium forms cesium hydroxide, in the Brunswick plant, 104 gm-mols of Cs(OH) is released to the suppression pool.

Sodium Pentaborate

After a LOCA, 1883 pounds of sodium pentaborate would be injected into the reactor vessel for reactivity control. It is assumed that this pentaborate eventually will get into the suppression pool water forming 5.095E-3 gm-mol boron/liter solution. By a buffering action, this solution would stabilize the pH of the suppression pool water.

Determination of Suppression Pool pH

The licensee calculated suppression pool pH values covering the period of 30 days after a LOCA. Two cases were considered:

Case 1 Release to the suppression pool of all acidic chemicals and cesium hydroxide
Case 2 Release to the suppression pool of all acidic chemicals and sodium pentaborate

In the first case, the values of suppression pool pH will decrease with time as the acidic chemicals will accumulate in the suppression pool water and eventually will reach a value well below 7.0.

In the second case the values of suppression pool pH will also decrease, but at a much slower rate. The lowest value of pH reached at 30 days after a LOCA will be 7.93.

The conclusion drawn from comparing these two cases is that without buffering action of the sodium pentaborate, it is not possible to maintain pH higher or equal to 7.0. Consequently, this will make the suppression pool water retain dissolved iodine. Therefore, sodium pentaborate will perform a dual function. In addition to its primary function of controlling reactivity of the reactor core after a LOCA, it will also control pH of the suppression pool water at the values required for retaining dissolved iodine.

In the appendix to the submittal addressing control pH in the suppression pool, the licensee determined the value of pH in a 30-day transient after a LOCA. The pH was determined by the amount of acidic and basic chemicals either released from the damaged core, or generated in the suppression pool by the radioactive environment existing after a LOCA. The licensee also considered the buffering effect of sodium pentaborate coming from the Standby Liquid Control System. The licensee concluded that without the buffering action of sodium pentaborate during the 30-day post-LOCA transient, the pH will decrease below 7.0. However, when the effect of sodium pentaborate is included the value of suppression pool pH will remain above 7.0, even without taking credit for cesium hydroxide. The NRC staff verified licensee's calculations of the type and amounts of chemicals released to the suppression pool water and concurs that they represent realistic values for the conditions existing in the post-LOCA environment in the Brunswick plant. The NRC staff also reviewed the methodology for determining the resultant suppression pool pH and found that it is based on the best available information providing realistic values. The NRC staff concludes, therefore, that the post-LOCA suppression pH, specified in the appendix to the licensee's submittal, represents a realistic estimate of the suppression pool pH which would exist in the suppression pool after a LOCA.

3.2.1.3 Containment Leakage Pathway

The Mark I primary containment is projected to leak at its design leakage rate of 0.5 percent of its contents by weight per day for the first 24 hours and then at 0.4 percent for the remainder of the 30-day accident duration. RG 1.183, Appendix A, Section 3.7 states that for BWRs, primary containment leakage may be reduced after the first 24 hours, if supported by plant configuration and analysis, to a value not less than 50 percent of the technical specification leak rate. In its letter dated February 4, 2002, CP&L responded to an NRC staff question on the basis for its assumed 20 percent leakage reduction after 24 hours. The licensee stated that although the BSEP post-LOCA primary containment pressure response shows a substantial reduction from the BSEP design pressure (P_a) of 49 psig by 24-hours post-LOCA, because of nitrogen inerting for combustible gas control, the primary containment pressure is expected to gradually increase over the 30-day LOCA event. CP&L based the reduction in primary containment leakage after 24 hours on the 30-day endpoint pressure of 31 psig. CP&L calculated a reduced leakage rate of 20 percent, based on a relationship that flow rate is proportional to the square root of the differential pressure. The NRC staff notes that this relationship is determined for flow through valves, fittings and pipes and does not find it applicable for calculating the reduction in primary containment leakage. However, considering the calculated reduction in BSEP post-LOCA primary containment pressure, the NRC staff does find the assumption of 20 percent reduction in drywell leakage after 24 hours to be acceptable at BSEP.

Leakage from the drywell / wetwell will collect in the free volume of the secondary containment and be released to the environment via ventilation system exhaust or leakage. Following a LOCA, the standby gas treatment system (SGTS) fans start and draw down the secondary

containment to create a negative pressure with reference to the environment. This pressure differential ensures that leakage from the drywell / wetwell is collected and processed by the SGTS. SGTS exhaust is processed through charcoal filter media prior to release to the environment via the site's elevated stack. CP&L does not credit dilution or holdup of leakage in the secondary containment. In addition, CP&L assumes that a positive pressure exists in the secondary containment for the first 5 minutes after the accident and that this leakage is released directly to the environment as a ground level release.

The original BSEP licensing basis did not consider a positive pressure and there are no secondary containment drawdown time surveillance requirements (SRs). CP&L has stated that existing SRs will detect significant increases in secondary containment leakage (SR 3.6.4.1.3) or degradation in SGTS performance (SR 3.6.4.3.1) prior to exceeding the assumed five-minute drawdown time. CP&L also noted that its assumption of 5 minutes is larger than the two-minute drawdown time typical at BWRs having this SR. CP&L has taken the position that the existing BSEP TS are adequate to ensure this analysis assumption is met. A similar position was found acceptable by the NRC staff in the review of the implementation of an AST at Duane Arnold, Amendment 240 to License No. DPR-49, dated July 31, 2001. While an explicit drawdown time specification would be preferable, the NRC staff feels that the position put forth by CP&L is acceptable. The NRC staff expects that the existing SRs will identify leakage that prevents achieving the requisite negative pressure and will adequately serve as an indication that the analysis assumption cannot be met. The NRC staff expects that CP&L's decision to not credit holdup and dilution in the secondary containment and to not credit wetwell free volume in establishing containment release rates largely compensates for any uncertainty in the drawdown time assumption.

3.2.1.4 Main Steam Isolation Valve Leakage

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

The NRC staff finds assumptions on credit for holdup and plate-out in the condenser and main steam lines acceptable based on the NRC staff's conclusion above in Section 3.1 of this SE that the main steam system piping and components that comprise the ALT System are seismically rugged and are able to perform the safety function of an MSIV leakage treatment system. CP&L assumes that the elemental iodine and particulate filter removal efficiency of the main condenser is 99.6 percent for the alternate leakage treatment pathway through the main steam drain lines downstream of the outboard MSIV to the isolated main condenser. This

efficiency was determined using a methodology in the approved BWROG licensing topical report NEDC-31858P-A, “BWROG Report for Increasing Main Steam Isolation Valve Leakage Rate Limits and Elimination of Leakage Control Systems,” dated August 1999. This methodology was based on the TID-14844 iodine species fractions of 91 percent elemental, 4 percent organic and 5 percent particulate. The NRC staff believes that the condenser iodine removal efficiency determined using that methodology is bounding for the AST with its 95 percent aerosol (particulate) iodine species and 0.15 percent organic iodine species.

For deposition in the main steam lines, CP&L uses the Brockmann-Bixler pipe deposition model incorporated in the NRC-sponsored RADTRAD computer code. CP&L conservatively modeled two of the four main steam lines with a total MSIV leakage of 46 scfh, which is equivalent to the maximum leakage allowed by BSEP TS. These flows are reduced by 20 percent at 24 hours to reflect the decrease in containment pressure—the driving force for the steam release. One of the two steam lines was designated as being faulted within the containment with its inboard MSIV assumed to have failed open. As a result, CP&L conservatively did not credit deposition between the RPV and the outboard MSIV in this faulted steam line. CP&L credited deposition downstream of the outboard MSIVs in both steam lines. The NRC staff considers this approach conservative in that it exceeds minimum regulatory guidance in that multiple failures are postulated.

Two aspects of CP&L’s use of the Brockmann-Bixler deposition model warrant discussion:

- Section 2.2.6.1 of NUREG/CR-6604, “RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation,” briefly describes the bases of the Brockmann-Bixler model. The deposition efficiency is defined in the context of “. . .the horizontally projecting lower surface. . . .” CP&L used plant-specific piping geometry data and included only horizontal main steam lines in determining the total volume and interior surface area used in the model.

- Appendix A to RG 1.183, §6.3, provides the guidance that steam line deposition models should be based on the assumption of well-mixed volumes, but other models such as slug flow may be used if justified. Brockmann-Bixler is a plug-flow model.

During a DBA LOCA, the flow pattern in the main steam line could be plug flow, well mixed, or some combination of the two. Plug flow effectively results in a longer fission product transport time and more deposition in the steam line. The NRC staff considered main steam line deposition in its review of the Perry pilot AST application. The NRC staff’s analysis is documented in the SE for that amendment and in the NRC staff technical report AEB-98-03, “Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term.” The conclusions of this report formed part of the basis of the cited RG 1.183 guidance. Appendix A of AEB-98-03 addresses the use of plug-flow and well-mixed models. The NRC staff also considered main steam line deposition credit in its review of the BWROG NEDC-31858P topical report. This model was based on plug flow in the main steam lines and main condenser. Use of this model did require the applicant to seismically qualify the steam lines and the main condenser. The NRC staff finds the use of plug flow to be an acceptable deviation from the guidance of RG 1.183, based on the existence of the seismically adequate alternative leakage paths and main condenser.

3.2.1.5 Leakage from Emergency Core Cooling Systems

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

The analysis considers the equivalent of 20 gpm ECCS leakage starting at the onset of the LOCA. This leakage is 20 times higher than the TS action limits. This conservatism includes a factor of 2 multiplier, in accordance with guidance in RG 1.183, to address increases in the leakage due to normal material degradation between surveillance tests, plus an additional factor of 10. CP&L assumes 10 percent of the iodine in the ECCS leakage becomes airborne and is available for release. This release is 97 percent elemental and 3 percent organic iodine. No credit was assumed for holdup and dilution in the secondary containment. As was assumed for the primary containment leakage pathway, the leakage enters the environment as an unfiltered ground level release for the first 5 minutes after the event starts. After this 5-minute positive pressure period, the leakage enters the environment via the SGTS as a filtered elevated release.

3.2.1.6 Offsite Doses

CP&L evaluated the maximum 2-hour TEDE to an individual located at the exclusion area boundary (EAB) and the 30-day TEDE to an individual at the outer boundary of the low populations zone (LPZ). The resulting doses are less than the 10 CFR 50.67 criteria.

3.2.1.7 Control Room Doses

CP&L evaluated the dose to operators in the control room and to personnel in the technical support center (TSC). It was assumed that the control room would be manually isolated at 20 minutes following the start of the event. The TSC is isolated manually, at 2 hours after the start of the event. Prior to the event, the control room ventilation system draws in 2100 cfm of outside air. Once isolation occurs, this intake flow is stopped and 1500 cfm of filtered makeup and 400 cfm of filtered recirculation are started. The control room emergency filters have efficiencies of 95 percent for particulates, 90 percent for elemental iodine, and 90 percent for organic iodine. The TSC normal outside air makeup is 4690 cfm. Once the TSC ventilation is isolated, 4690 cfm of outside air passes through the TSC filters. The TSC filters have efficiencies of 95 percent for particulates, 90 percent for elemental iodine, and 90 percent for organic iodine.

Although the TSC and control room are both designed to be pressurized during an accident, CP&L assumes that unfiltered inleakage occurs. Since this inleakage has not been quantified, CP&L analyzed three cases; 10,000 cfm, 3,000 cfm, and 0 cfm unfiltered inleakage for the control room and 10,000 cfm, 1720 cfm, and 0 cfm unfiltered inleakage for the TSC. The results of the sensitivity analysis indicated that the higher doses are associated with the 10,000 cfm unfiltered inleakage rate. The unfiltered inleakage is assumed to commence at the onset of the event and continue for 30 days for both the control room and the TSC.

The NRC staff is currently developing regulatory guidance regarding control room habitability, including surveillance testing of unfiltered inleakage. In addition, the Nuclear Energy Institute has developed an industry initiative document on control room habitability: NEI 99-03, "Control Room Habitability Assessment Guidance." The NRC staff's acceptance of CP&L's unfiltered inleakage assumption here does not foreclose on any future generic regulatory actions that may become applicable to BSEP in this regard.

CP&L analyzed the control room and TSC doses over a 30-day period. The resulting 30-day TEDE to an individual in the control room or TSC is less than the 10 CFR 50.67 criteria.

3.2.1.8 LOCA Conclusion

Based on it's review discussed above, the NRC staff concluded that the licensee's application of the AST to the BSEP LOCA analysis is acceptable. Table 1 provides the analysis assumptions found acceptable by the NRC staff. Table 2 provides the doses projected by CP&L.

3.3 Main Steam Line Break (MSLB)

The accident considered is the complete severance of a main steam line outside the primary containment with the reactor operating at 2923 MWt. The radiological consequences of a break outside containment will bound the results from a break inside containment. The MSIVs are assumed to isolate the leak within 5.5 seconds. This assumed time is based on the allowed MSIV closure time of 5 seconds and the response time for the isolation logic of 0.5 seconds. The analysis is performed for two activity release cases, based on the maximum equilibrium and pre-accident iodine spike concentrations of 0.2 μCi/gm and 4 μCi/gm dose equivalent I-131, respectively. All of the accident activity was assumed released within 5.5 seconds following the accident as a ground level release, with no credit for turbine building holdup or dilution. These assumptions are in accordance with RG 1.183 guidance.

The control room ventilation system was modeled as described above for the LOCA. The MSLB analysis demonstrates that the 30-day control room dose is nearly insensitive to the time of control room isolation, and the licensee reported the bounding dose that encompasses both isolated and unisolated conditions for the control room. The MSLB control room dose results also correspond to an assumed unfiltered inleakage rate of 0 cfm. The licensee's sensitivity analysis demonstrated that the control room doses are maximized at zero unfiltered inleakage due to the essentially instantaneous nature of the release (5.5 seconds) and the effect of the higher inleakage of less-contaminated air following the release that sweeps out the contaminated air within the control room envelope. These assumptions are conservative.

Based on this review, the NRC staff concluded that the licensee's application of the AST to the BSEP MSLB analysis is acceptable. Table 1 provides the analysis assumptions found acceptable by the NRC staff. Table 2 provides the doses projected by CP&L.

3.4 Control Rod Drop Accident (CRDA)

This accident analysis postulates a sequence of mechanical failures that result in the rapid removal (i.e., drop) of a control rod. Localized damage to fuel cladding and a limited amount of fuel melt are projected. A reactor trip will occur. The MSIVs are assumed to remain open for the duration of the event. CP&L has projected that 1200 fuel rods would be breached by the event, and of these damaged rods, 0.77 percent would exceed the threshold for melting.

The CRDA analysis was performed using the gap fractions and fuel melt fractions from Appendix C of RG 1.183. Ten percent of the core inventory of noble gases and iodines is assumed to be in the fuel gap. For melted fuel, 50 percent of the iodines and 100 percent of the noble gases are assumed released instantaneously to the reactor coolant.

It is assumed that 100 percent of the noble gases but only 10 percent of the iodines released reach the main condenser due to plate-out in the RPV and main steam lines. Of the iodine that enters the main condenser, 90 percent plates out. 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 over 24 hours as a ground level release. The control room ventilation system was modeled as described above for the LOCA.

Based on this review, the NRC staff concluded that the licensee's application of the AST to the BSEP CRDA analysis is acceptable. Table 1 provides the analysis assumptions found acceptable by the NRC staff. Table 2 provides the doses projected by CP&L.

3.5 Other Radiological Consequence Analyses

CP&L considered the impact of the increased thermal power and AST on the equipment qualification as part of the separate extended power uprate submittal. Consistent with NRC staff practice (SECY-99-240), CP&L used the TID-14844 source term in examining these potential radiological impacts on equipment qualification.

CP&L evaluated the potential impacts on compliance with various BSEP commitments to NUREG-0737 and performed calculations to update the post-accident access for NUREG-0737 Item II.B.2 and direct shine doses in the control room using the proposed uprated power and the AST. CP&L evaluated the impact on the post-accident vital mission doses associated with Post-accident Sampling System (PASS) sample collection and SGTS stack sampling. The calculations address the effects on mission dose evaluations and control room dose calculations from revised post-accident cloud immersion, reactor building shine and reactor coolant / drywell atmosphere contained source doses. The evaluation results indicate that the BSEP plant shielding is adequate to maintain all post-accident vital mission doses within regulatory criteria. Based on the information provided by the licensee, the NRC staff finds these evaluations meet the guidance of RG 1.183 and are therefore acceptable.

3.6 Atmospheric Relative Concentration Estimates

3.6.1 Meteorological Data

CP&L calculated new relative concentration (X/Q) estimates for the LOCA, MSLB and CRDA dose assessments described above using onsite meteorological data collected during calendar years 1996 through 1999. These data were measured at 11.5 and 104.6 meters above grade at the BSEP site. The licensee confirmed that the meteorological measurement program meets the recommendations in RG 1.23, "Onsite Meteorological Programs." The licensee states that the tower area is free of natural vegetation and man-made structures that might otherwise influence meteorological measurements. Scheduled calibrations are performed on a semi-annual basis and wind and temperature instruments are replaced with systems traceable to National Institute of Science and Technology sensors. Data are accessed offsite daily and compared with other local observations. Plant personnel also log observations once every 12 hours. The onsite meteorological measurement system receives additional data observation because it is included in the scope of the Maintenance Rule. CP&L notes that the BSEP meteorological measurement system has maintained a data recovery rate of greater than 90 percent since a new system was installed in 1996. This includes years with data loss due to the passage of three hurricanes.

The NRC staff performed a review of the meteorological data submitted by CP&L using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data." Further review was performed using a computer spreadsheet. In the SE supporting Amendments 218 and 244 to Facility Operating License Nos. DPR-71 and DPR-62, respectively, issued on March 14, 2002, the NRC staff previously stated that it had confirmed that the 1996 through 1999 joint recovery rate of the submitted wind speed, wind direction and atmospheric stability data used in the X/Q estimates was greater than 90 percent. However, subsequent examination of the data revealed some occurrence of wind data remaining unchanged for two or more consecutive hours. When these frequencies were combined with the estimated frequencies of invalid data, NRC staff could no longer confirm the licensee's estimates of data recovery. However, even with the uncertainty, the NRC staff estimates that the recovery is still, at a minimum, near 90 percent and the data recovery uncertainties should not have a significant impact on the licensee's X/Q estimates. With respect to atmospheric stability measurements, the length and time of occurrence of stable and unstable atmospheric conditions appeared reasonable, although the reported occurrence of stability class A (extremely unstable) in 1998 was considerably lower than for the other 3 years of the measurement period. This may be due, in part, to the relative paucity of atmospheric stability data submitted for the month of June when a high occurrence of stability class A conditions might be expected during daytime hours. Reported stability class occurrence as a function of time of day was very good. Stable and neutral conditions were consistently reported to occur at night and unstable and neutral conditions during the day. The longest occurrence of any single unstable category was 8-consecutive hours which is consistent with expected meteorological conditions. Reported wind data indicate a relatively high occurrence of calm wind speeds at the lower level with some year-to-year variability in frequency of occurrence of faster winds at both levels. Wind direction frequency occurrence at the lower level was very similar from year to year. Upper level wind direction frequency occurrence was more variable and, as a result, the year-to-year correlation between the two levels was also somewhat variable. However, both heights showed distinct bimodal flow, from the southwest and generally northeasterly quadrants.

3.6.2 Exclusion Area Boundary and Low Population Zone Relative Concentration Estimates

The licensee calculated X/Q values for the Exclusion Area Boundary (EAB) and Low Population Zone (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." The licensee made calculations for an EAB distance of 3000 feet and LPZ distance of 2 miles. Values were calculated for both ground level and elevated releases. Since the Brunswick site is more than 2 miles from the Atlantic coastline, for elevated releases, fumigation was assumed to occur during the ½-hour interval that would result in the highest dose.

3.6.3 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. New values for various combinations of postulated release and receptor locations for both units were calculated and the highest values for each category were selected as inputs to the dose calculations. The limiting cases were determined using minimum distances from the closest wall of the reactor building, reactor building exhaust vent, turbine building exhaust, condenser location within the turbine building, and offgas stack. All releases were assumed to be ground level point releases except those from the offgas stack. The offgas stack was considered to be an elevated point release with fumigation assumed to occur during the ½-hour interval that would result in the highest dose. For non-fumigation conditions, CP&L made calculations using the ARCON96 methodology from the offgas stack to the control room intakes. These values were compared with calculations made for the ground level concentration resulting from an elevated release using the PAVAN methodology. The higher of the two sets of values, those from the PAVAN calculations, were used in the dose estimates.

3.6.4 Conclusion Regarding New X/Q values

The NRC staff qualitatively reviewed the inputs to the PAVAN and ARCON96 codes and found them to be consistent with site configuration drawings and other information in the BSEP UFSAR and staff practice. Based on this review, the NRC staff finds the new X/Q values acceptable.

3.7 <u>Proposed Technical Specification Changes</u>

CP&L requested changes to some TS, and some conforming changes to TS Bases. The following TS are affected by the proposed changes:

a. In Section 1.1, the definition of "Dose Equivalent Iodine-131," is changed to substitute Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors," 1989, and FGR-12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993, for the existing reference. The word "thyroid," was deleted.

The existing definition is based on the dose conversion factors provided in TID-14844. This change conforms to the implementation of the AST. The new citations are as cited in RG 1.183 and are acceptable. These revised values were used in the re-analysis of the MSLB accident, found acceptable above. No other DBA analysis, including the FHA approved in Amendment No. 218 to Facility Operating License No. DPR-71 and Amendment No. 244 to Facility Operating License No. DPR-62, is affected by this change. The change is acceptable.

b. Various references to 10 CFR Part 100, in both the TS and their bases, will be changed to 10 CFR 50.67, to conform to the AST implementation.

With the implementation of the AST, the accident dose guidelines of 10 CFR Part 100 are superseded by the dose criteria in 10 CFR 50.67. The whole body and thyroid doses of 10 CFR Part 100 are replaced by the TEDE criteria of 10 CFR 50.67. This is a conforming change. The analyses performed in support of this amendment (and that for the FHA in Amendment No. 218 to Facility Operating License No. DPR-71 and Amendment No. 244 to Facility Operating License No. DPR-62) determined radiological consequences in terms of the TEDE dose quantity, and were shown to be in compliance with the dose criteria in 10 CFR 50.67. The changes are acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 change the SRs. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (66 FR 46477). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

In its application dated August 1, 2001, CP&L had proposed a full scope implementation of the AST. Amendment No. 218 to Facility Operating License No. DPR-71 and Amendment No. 244 to Facility Operating License No. DPR-62, approved a selective implementation of AST for the FHA in support of requested changes to TS during movement of irradiated fuel, were issued on March 14, 2002. With this SE, the NRC staff completed its review of the remaining DBAs and other radiological consequences of an AST implementation and has determined that CP&L has met the requirements of 10 CFR 50.67 and the guidance of RG 1.183 for a full-scope implementation.

The NRC staff reviewed the assumptions, inputs, and methods used by CP&L to assess the radiological impacts of the proposed changes. In doing this review, the NRC staff relied upon information placed on the docket by CP&L, NRC staff experience in doing similar reviews and, where deemed necessary, on NRC staff confirmatory calculations. The NRC staff finds that CP&L used analysis methods and assumptions consistent with the conservative guidance of RG 1.183, the proposed TS changes, and the extended power uprate. The NRC staff compared the doses estimated by CP&L to the applicable criteria and to the results of confirmatory analyses by the NRC staff. The NRC staff finds, with reasonable assurance, that the licensee's projections of the EAB, LPZ, control room, and TSC total effective dose equivalent due to postulated DBAs at BSEP will comply with the requirements of 10 CFR 50.67 and the guidance of RG 1.183. The NRC staff finds reasonable assurance that BSEP AST implementation will continue to provide sufficient safety margins with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression and in analysis assumptions and parameters.

Since these analyses were performed at a power level of 2981 MWt (102 percent of 2923 MWt), the NRC staff finds that the radiological consequences of these DBAs would remain bounding up to a rated thermal power of 2923 MWt. However, the approval of this amendment does not constitute authority to operate above the current licensed rated thermal power of 2558 Mwt. Operation at an extended power level is addressed in a separate amendment request dated August 9, 2001, still under review.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the BSEP design basis is superseded by the AST proposed by CP&L in its application of August 1, 2001, as supplemented. The previous offsite, control room, and TSC accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR 50.67 or small fractions thereof, as defined in RG 1.183. All future radiological analyses performed to demonstrate compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as described in the now-updated BSEP design basis.

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.

7.0 REFERENCES

1. Letter from Carolina Power & Light Company to NRC regarding license amendment request to adopt alternative radiological source term, dated September 27, 2001.

2. Letter from Carolina Power & Light Company to NRC regarding license amendment request to adopt alternative radiological source term, dated January 24, 2002.

3. General Electric Nuclear Energy, September 1993, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," NEDC 31858P, Rev. 2.

4. Safety Evaluation of GE topical report, NEDC-31858P, Rev. 2, “BWROG report for increasing MSIV leakage limits and elimination of leakage control systems,” March 3, 1999.

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Date: May 30, 2002

TABLE 1

RADIOLOGICAL ANALYSIS ASSUMPTIONS (pg. 1 of 4)

Assumptions Common to One or More Analyses

Reactor power, MWt (102% of 2923)	2981
RCS specific activity, equilibrium µCi/gm dose equivalent I-131	0.2
RCS specific activity, spike µCi/gm dose equivalent I-131	4.0
Dose conversion factors	FGR11 and FGR12
Control Room volume, ft^3	298,650
Control Room manual isolation, minutes	20
Control Room normal ventilation makeup flow, cfm	2100
Control Room filtered makeup flow, cfm	1500
Control Room filtered recirculation flow, cfm	400
Control Room filter efficiency, %	
Aerosol 95	
Elemental 90	
Organic 90	
Control Room unfiltered inleakage, cfm	0, 3,000, and 10,000
Control Room breathing rate, m^3/sec	3.47E-4
Control room occupancy factors	
0-24 hours	1.0
1-4 days	0.6
4-30 days	0.4

Control Room χ/Q, sec/m^3

Period	Elv Stack	TB Exhaust	Condenser	RB Wall	RB Vent
0-2 hrs	4.65E-6	5.55E-4	1.15E-3	4.05E-3	1.48E-3
2-8 hrs	1.73E-6	4.41E-4	1.03E-3	3.67E-3	1.30E-3
8-24 hrs	1.06E-6	2.21E-4	4.90E-3	1.74E-3	6.74E-4
1-4 days	3.62E-7	1.54E-4	4.11E-3	1.44E-3	4.90E-4
4-30 days	7.76E-8	1.21E-4	2.63E-4	1.02E-3	3.49E-4
Fumigation	3.19E-4				

TSC volume, ft^3	208,800
TSC manual isolation, hours	2
TSC normal ventilation, cfm	4690
TSC filtered makeup, cfm	4690
TSC filtered recirculation, cfm	37990

TSC filter efficiency, %	
Aerosol	95
Elemental	90
Organic	90
TSC occupancy factors	
0-24 hours	1.0
1-4 days	0.6
4-30 days	0.4

TSC χ/Q, sec/m^3

Period	Elv Stack	Condenser	RB Wall
0-2 hrs	3.74E-6	2.04E-4	3.11E-4
2-8 hrs	1.25E-6	1.90E-4	2.55E-4
8-24 hrs	7.23E-7	9.89E-5	1.15E-4
1-4 days	2.21E-7	6.96E-5	8.57E-5
4-30 days	4.01E-8	4.88E-5	5.52E-5
Fumigation	2.16E-4		

EAB χ/Q, sec/m^3

Period	Ground	Elv Stack
0-2 hrs	2.20E-3	3.63E-6
0-8 hrs	1.23E-3	2.04E-6
8-24 hrs	9.26E-4	1.53E-6
1-4 days	4.96E-4	9.27E-7
4-30 days	2.02E-4	4.80E-7
Fumigation	5.85E-5	

LPZ χ/Q, sec/m^3

Period	Ground	Elv Stack
0-2 hrs	7.77E-4	3.50E-6
0-8 hrs	3.36E-4	1.92E-6
8-24 hrs	2.21E-4	1.42E-6
1-4 days	8.90E-5	7.40E-7
4-30 days	2.41E-5	3.07E-7
Fumigation	1.88E-5	

Assumptions for LOCA Analyses

Core inventory	Calculated by ORIGEN
Onset of gap release phase, min	2.0
Core release fractions and timing–drywell atmosphere	RG 1.183, Table 1

Core release fractions and timing–ECCS leakage

Duration, hrs	0.5	1.5
Iodine: 0.05	0.25	

TABLE 1 (pg. 3 of 4)

	Atmosphere	Suppression Pool
Iodine species fraction		
Particulate/aerosol	95	0
Elemental	4.85	100
Organic	0.15	0
Drywell volume, ft^3		164,000
Containment release, %/day		0.5
After 24 hours		0.4
MSIV Leakage (total) scfh		46
Duration of release, days		30
Reactor building positive pressure, minutes		5
Drywell natural deposition		10% Powers Model
Main steam line deposition model		Brockmann-Bixler
Pressure (0-30 days), atm		4.33
Temperature (0-30 days), degrees F		560
Condenser deposition elemental & particulate efficiency, %		
Main steam primary drain line path		99.8
Main steam alternate drain line path		99.6
SGTS filter efficiency, all species, percent		99
ECCS leak rate, gpm (includes 20x multiplier)		20
Duration of release, days		30
Suppression pool liquid volume, ft^3		86,450

Assumptions for MSLB Analyses

Mass release	
Steam, lbm	9,450
Liquid, lbm	37,600
Break isolation time, sec	5.5
RCS activity	
Equilibrium iodine spike case	0.2 µCi/gm D.E.I-131
Pre-incident iodine spike case	4.0 µCi/gm D.E.I-131
Iodine species release fraction to environment	
Elemental	0.97
Organic	0.03

TABLE 1 (pg. 4 of 4)

Assumptions for CRDA Analyses

Core inventory	Calculated by ORIGEN
Radial peaking factor	1.5
Fuel bundles in core	560
Fuel rods in bundle (effective)	87.33
Rods that exceed DNB	1200
Fraction of rods that exceed DNB that experience melt	0.0077
Gap fraction, noble gas and iodine	0.01
Melt isotopic composition	
Noble gases	1.0
Iodine	0.5
Fraction of core release that enters condenser	
Noble gases	1.0
Iodine	0.1
Iodine retention in condenser	0.9
Iodine species release fraction to environment	
Elemental	0.97
Organic	0.03
Condenser leakage, %/day	1.0
Release duration, hours	24
Turbine building ground level release	
Control room unfiltered inleakage, cfm	0

TABLE 2

RADIOLOGICAL ANALYSIS RESULTS,[1] REM TEDE

Event	0-2 hr EAB	30-day LPZ	30-day CR	30-day TSC
Loss-of-Coolant Accident	0.64	1.36	3.62	1.15
Main Steam Line Break				
Equilibrium Activity	0.127	0.045	0.025	(2)
Pre-incident Spike	2.52	0.89	0.5	(2)
Control Rod Drop Accident	0.27	0.22	0.28	(2)

Notes

1. Determined by CP&L and confirmed by NRC staff.
2. Not calculated. Control room doses are limiting based on location of release in relation to control room and TSC intakes.

TABLE 3

ACCIDENT DOSE ACCEPTANCE CRITERIA,[1] REM TEDE

Event	0-2 hr EAB	30-day LPZ	30-day CR	30-day TSC[2]
Loss-of-Coolant Accident	25	25	5	5
Main Steam Line Break				
Equilibrium Activity	2.5	2.5	5	5
Pre-incident Spike	25	25	5	5
Control Rod Drop Accident	6.3	6.3	5	5

Notes

1. From RG 1.183, Table 6 and 10 CFR 50.67.
2. RG 1.183, subsection 4.4 states that acceptance criteria for various NUREG-0737 items should be updated for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii).